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3.0 Design Criteria - Structures, Components, Equipment and Systems

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.

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3.1 Conformance With General Design Criteria

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: 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, Table 3-5, and Table 3-6. 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, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

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

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, reliable sources of either water or carbon dioxide are provided to appropriate parts of the station.

Reference: Section 9.5.1

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

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 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.

Class 1E electrical equipment is designed and qualified to perform its safety function(s) under the harsh environmental conditions applicable to its location.

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 2, Chapter 3, and Chapter 6

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

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 no 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, and 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 (manually or automatically by controlling thermal load).
2. Reactor coolant temperature (manually or automatically by rod control cluster assembly motion, in sequential groups).
3. Reactor coolant pressure (manually 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 (manually by full length control 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 ensure 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 indicators 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 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

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

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 over-pressure 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: 3.6, 3.7; and Chapter 5.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

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

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 effects 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 overpressure 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, 6.3, and 6.7

CRITERION 17 - ELECTRICAL POWER SYSTEMS

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

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

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

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

DISCUSSION: An onsite and an offsite power system are provided for each unit with sufficient capacity and capability to power those systems and components required for safety.

Reliability of offsite power to the station is assured by six independent double-circuit connections between the 230kV switchyard and the Duke Transmission System and two separate and independent transmission lines per unit connecting the switchyard to the station. These two lines per unit supply power to two half-sized main stepup transformers which reduce the voltage to 20.9kV. The use of two generator circuit breakers per unit allows immediate access to each of the preferred power sources. These sources maintain their independence within the auxiliary power system through separate voltage transformations from 20.9kV to 6.9kV and then to 4.16kV. At the 4.16kV level these sources connect to and supply the Essential Auxiliary Power System.

The onsite electric power supplies, including the two 7000 KW diesel generators per unit, the four 125VDC vital batteries per unit and their associated distribution systems, have sufficient independence, redundancy, and testability to perform their safety function assuming a single failure. The 4.16kV essential system supplies those systems and components required for safety. The 125V Vital DC System consists of four independent load groups each provided with a battery and a battery charger. This system supplies the vital instrumentation and control load required for safety. The specific criteria used in the design of the Class 1E power systems is in accordance with IEEE 308-1971.

Reference: Chapter 8

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

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

DISCUSSION: Provisions are made for periodic testing of all important components of the Essential Auxiliary Power System. Further provision is made for periodic testing of the emergency diesel generators to assure their capability to start and to accept loads within design limits. Electric power systems important to safety are designed to allow periodic testing to the extent practical. Staggered tests are employed to avoid the testing of redundant equipment at the same time.

The 230kV switchyard power circuit breakers, the generator circuit breakers, and their associated protective relaying are inspected, tested, and maintained on a routine basis. The 13.8kV, 6.9kV, and 4.16kV circuit breakers and associated equipment are tested in-service by opening and closing the breakers so as not to interfere with the operation of the unit. The 600-volt breakers, motor contactors, and associated equipment are also tested in-service by opening and closing the breakers and contactors so as not to interfere with unit operation. Additionally, the protective relaying associated with the 13.8kV, 6.9kV, 4.16kV, and 600-volt power systems is inspected, tested, and maintained on a routine basis.

Reference: Chapter 8

CRITERION 19 - CONTROL ROOM

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

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

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 shutdown condition for an extended period of time from locations outside the main control room. 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 are not subjected to doses under postulated accident conditions which would exceed 5 rem total effective dose equivalent (TEDE).

Reference: Chapter 7 and Chapter 12 and Sections 3.8, 6.4, and 9.1

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The Protection System shall be designed:

1. To initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and
2. To sense accident conditions and to initiate the operation of systems and components important to safety.

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 279-1971. The Reactor Protection System automatically initiates a reactor trip when any monitored 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 15.1 for ANS N18.2, Conditions I and II.

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

DISCUSSION: The Protection System is designed to comply with the intent of IEEE-27 9-1971, "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 under voltage 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. It also utilizes redundant analog channels measuring the same parameter and two redundant logic trains, either of which will actuate safety injection and/or containment spray.

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 principles 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, Chapter 7, and Section 3.11

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

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

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 deenergize-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

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

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 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 15.1. 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 REDUNDANCY AND CAPABILITY

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

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 condition irrespective 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 facility procedures. The Core Operating Limits Report required by the Technical Specifications 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 Section 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, and Chapter 15

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.

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.

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

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 in-service testing.

Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradation and post-irradation 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 leaktight integrity, and (2) an 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 Boiler and Pressure Vessel 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, "Recommended 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 being 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 or 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 preset 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 the 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 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 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 I 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 System in this function. The systems together accommodate the single-failure criterion.

Reference: Chapter 15, and Sections 5.4.7 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 Criteria for Emergency Core Cooling System for Light-Water Power Reactors of December 1973 (10 CFR 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 fuel-clad 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 the 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 code.

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

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 test 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, valve 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: Section 15.1 and Section 6.3

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 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: 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: Sections 6.2.2, 6.5 and 6.7

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

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

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. The 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, 6.5 and 6.7

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM.

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

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 periodically 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, 6.5, and 6.7

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP.

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

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

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. It has the capability of cross-connection with another identical system.

Hydrogen control within the containment is provided by redundant electrical hydrogen recombiners. Hydrogen pocketing in subcompartments of the containment is prevented by use of the Containment Air Return and Hydrogen Skimmer System.

Reference: Sections 6.2, 6.5 and Chapter 15

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS.

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

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

Reference: Sections 6.2 and 6.5

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS.

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

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: Sections 6.2 and 6.5

CRITERION 44 - COOLING WATER

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

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

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 operation 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: Section 9.2

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

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

DISCUSSION: The Component Cooling System (CCS) and the Nuclear Service Water 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 Nuclear Service Water Pump House thus allowing proper maintenance and inspection.

Duke Power Company'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

CRITERION 46 - TESTING OF COOLING WATER SYSTEM.

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

DISCUSSION: The Component Cooling System (CCS) and the Nuclear Service Water System (NSWS) are in operation during normal operation or shutdown. The structural and leak-tight integrity of the CCS and NSWS and the operability and performance of active components are continually 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: Section 9.2 and Technical Specifications.

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 as required by §50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of 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 cooling capacity of either the Containment Ventilation System and/or the Containment Spray System is adequate to prevent over-pressurization of the structure, and to return the Containment to atmospheric pressure.

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 preoperational integrated leak rate testing of the Containment and an inleakage rate test 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 Power Company'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 TESTING AND INSPECTION.

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

DISCUSSION: The Containment is designed such that integrated leak rates can be run during unit lifetime, and penetrations which have resilient seals may be leak tested at design pressure at any time. Bellows on mechanical penetrations are designed for a structural integrity test at reduced pressure.

Reference: Section 6.2

CRITERION 54 - 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, 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 containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

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

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 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 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 of radioactive wastes generated during normal operation.

Chapter 11 describes the Waste Processing System, the 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 designed 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, 11.4 and 12.3

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING.

Criticality in the Fuel Storage and Handling System shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

DISCUSSION: During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at a level sufficient to shutdown the core at a Keff ≤ 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 10CFR 50.68(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 maintained 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 11.5

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES.

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

DISCUSSION: The containment atmosphere, the unit vent, and the monitor tank building gaseous effluent, as well as the 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, and ECCS pump rooms) 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.

Reference: Sections 11.5 and 12.3

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3.2 Classification of Structures, Systems and Components

Note:

This section of the UFSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document (DBD) for this system/structure.

The classification of mechanical structures, systems, and components is based upon the requirements specified in 10CFR50.55a, GDC 1.14, and ANS N18.2-1973(1975 Addenda). The four regulations (ANS N18.2 is a standard) require provisions for the design, construction, and operation of plant equipment to a quality level commensurate with the importance of the safety function each must perform. Codes and standards are used to assure the quality of the various aspects of design, construction, and oepration, and they are applied based on the assignment of the structures, systems, and components to appropriate classifications.

The principal areas requiring classification relate to seismic considerations, quality assurance level, safety function considerations, and which code or standard applies. The following paragraphs outline how Duke Power Company has met the requirements in each area.

The term “nuclear safety related” or “safety related” is used throughout Duke 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 A, B, and C (defined below). Tables 3-1, 3-2, 3-4, and the Catawba NUREG-0588 submittal (H.B. Tucker letter to H.R. Denton dated February 8, 1984) list the principal system components considered to be “Safety Related”.

Instrumentation and electrical component classification and standards are discussed in Section 7.0 and Section 8.0.

3.2.1 Seismic Classification

Those structures, systems, and components necessary to perform the functions established in the regulatory position 1a. through 1q. of Regulatory Guide 1.29 have a seismic classification of Category I. Duke Power Company has further assigned a classification of Seismic Category II to those structures, systems, and components whose continued function is not required during a seismic event, but whose failure could reduce the functioning of any Category I structure, system, or component, as described in position 2 of Regulatory Guide 1.29. Quality Assurance Level 4 has been identified and is assigned to non-safety Seismic Category II structures, systems, and components. Seismic Category III is assigned to structures, systems, and components for which there are no nuclear seismic design criteria (non-seismic), but seismic analysis may have been completed for other regulatory/code purposes. Seismic category or Safety Class assignments are listed in Table 3-1, Table 3-2, and Table 3-4 .

Catawba Nuclear Station is in compliance with Regulatory Guide 1.29.

3.2.2 Mechanical System Quality Group Classification

Regulatory Guide 1.26 defines four Quality Groups, A through D, to further clarify classifications required by 10CFR50.55a. Regulatory Guide 1.26 further associates each quality group with a specific ASME or ANSI code for quality criteria. ANSI N18.2-1973(1975 Addenda) defines four Safety Classes. Class 1 through Class 3 and NNS, to associate various ASME and ANSI codes

and standards with equipment design, construction, and operation. Table 3-3 lists the four quality groups and four Safety Classes to show their relationship to the various codes.

Duke Power has established Safety Class A through H for the classification of components, to faciiltate the assignment of code requirements for design and procurement purposes. Duke Class D is not used at the Catawba Nuclear Station because no application of this class is used in any system design, and is thus not defined herein. The Duke Safety Classes are listed in Table 3-5 and 3-6 with both Regulatory Guide 1.26 Quality Groups and ANSI N18.2-1973 (1975 Addenda) Safety Classes to show their relationships. To simplify processes for design, purchasing, installation, and other quality assurance processes, Duke has assigned most devices to safety classifications using the ANS N18.2 Class 1,2,3, NNS system and most pipe and valves using the Duke Safety Class A-H classification system.

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

> Duke Safety Class E and F piping and components meet the requirements of C.3, where Duke Class E is normally only applied to certain portions of radioactive waste management system, and Duke Class F is normally assigned to non-safety Seismic Class II portions of systems.
>
> The failure of a Duke Class E component or piping would not result in an adverse effect on the health and safety of the public since it 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, portions of gaseous waste management systems containing gases which are not normally held up for decay prior to release, and other tritiated, degassed water lines.
>
> Duke Class F is assigned to those systems and components whose failure could result in flooding to nuclear safety related components, or whose failure could jeopardize nuclear safety related piping or components during a safe shutdown earthquake.
>
> Duke Class E and F components and materials shall meet the code and standard requirements for Quality Code D, though manufacturers furnishing Class E components or materials are not required to have a quality assurance program meeting Appendix B of 10CFR50.

The hierachy of classifications is discussed in the following paragraphs, and begins with the four NRC Quality Groups, paraphrased below from Regulatory Guide 1.26:

Quality Group A: Those mechanical systems and components of the reactor coolant pressure boundary to be designed, fabricated, erected, and tested to the highest available national standards

Quality Group B: Those water and steam containing pressure vessels, heat exchangers, storage tanks, piping, pumps and valves that are not part of the reactor coolant pressure boundary defined in 10CFR50.2 but part of emergency core cooling, post accident heat removal, post accident fission product removal, or designed for reactor shutdown and/or residual heat removal.

Quality, Group C: Those water, steam, and radioactive waste containing systems/components providing cooling and sealing water for ECCS components and auxiliary feedwater. Also included are systems/components which may contain radioactive material and whose postulated failure would result in potential offsite doses in excess of 0.5 rem.

Quality Group D: Those water and steam containing components not part of the reactor coolant pressure boundary or included in quality groups B or C, but part of systems or portions of systems that contain or may contain radioactive material.

3.2.2.1 Safety Class Definitions

The Duke Class A, B, C, E, F, G, & H and ANS Safety Class levels 1, 2, 3, NNS have similar definitions as follows:

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

A detailed description of the reactor coolant pressure boundary is given in Section 5.0.

Duke Class B, ANS 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 coolatn 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 secondary side of the steam generator and including the outermost containment isolation valves.

Duke Class C, ANS 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, excluding the cleanup loop.
    e. On-site emergency power supply and support systems.
    f. Air clean-up systems other than those listed in Section 3.e. of Safety Class 2.
2. Non-safety 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.

c. Portions of the boron recovery system.

Duke Class E, ANS Safety Class - applies to piping and components which carry radioactive fluids whose failure would not result in exposures greater than the limits of 10CFR100. Components in this class do not require seismic qualification and have no safety functions. Components in this class are found in the Catawba radioactive waste systems.

Duke Class F, ANS Safety Class - applies to piping and components which have no safety Function, but because of their proximity to piping or components which do have a safety function, require seismic restraints. Components in this class are found in numerous systems especially in high density areas where system components are in close proximity.

Duke Class - applies to piping and components used on the secondary side of the plant, neither a safety function nor a requirement for seismic restraint. They are procured per ANSI B31.1 standards, but are designed and installed per industrial standard practices rather than nuclear standards. Components in this class are found in the turbine and re-heater systems of the secondary side of the plant.

Duke Class - applies to piping and components for which there is no NRC recognized wide standard, and are thus designed and installed per Duke Power developed specifications, or Duke Power specified industry standards. Components in this class are found in ventilation systems, conventional drain systems, and the fire protection system.

3.2.2.2 System Piping Classification

System piping is assigned to one of the classification discussed in 3.2.2.1, depending on the required function of the system or portion of a system, and as required to distinguish analysis and purchasing requirements. The classification of each pipe segment is shown on the system Flow Diagrams with appropriate symbology, and is shown with its code criteria in Table 3-5.

Piping of different safety classification is isolated by valving or orificing as required in the 1975 Addenda to ANSI N18-2-1973.

3.2.2.3 System Valve Classification

System valves are assigned to one of the classifications discussed in 3.2.2.1, depending on the required function of the system or portion of a system, and as required to distinguish analysis and purchasing requirements. Valves in a given run of piping are assigned to the same classification as the piping it is located in. The valve classifications and associated code criteria are shown in Table 3-6.

Transition between piping design parameters (pipe specification, design pressure and temperature, class, and material type) is typically accomplished at a valve, and the valve is included in the higher or most conservative of each parameter.

3.2.2.4 System Classification Identification

Detailed discussions of system/component classifications, code application, and compliance are included in Section 5.0, Section 6.0, Section 10.0, and Section 11.0.

Piping and valve classifications are shown on the system flow diagrams for the systems described in these sections, using the symbology shown on Figures 1-22, 1-23 and 1-24.

3.2.2.5 Criteria for Designation of Mechanical Systems and Components

The criteria used in determining nuclear safety-related mechanical systems and components are discussed in Section 3.2.2. Piping and specific valves in mechanical systems are not classified in Table 3-4. Instead, system flow diagrams are used to indicate the Duke piping classification. All piping and equipment in classifications A, B, C and D are considered nuclear safety-related. All penetrations are Class B except as discussed in Note 21, Table 3-4 Classification F is not nuclear safety-related and is applicable to items that are not classified as nuclear safety-related but which could reduce the ability of a nuclear safety-related structure, system, or component to perform its intended function. Piping and equipment in classification E, G, and H are considered non safety-related. Supports for piping and equipment follow the classification of the piping/equipment supported. In some cases safety-related supports are used with non-safety related piping/equipment.

3.2.3 Criteria for Designation of Structures

Structures are considered nuclear safety-related if necessary to assure:

1. The integrity of the reactor coolant pressure boundary,
2. The capability to shut down the reactor and maintain it in a safe shutdown condition, or
3. The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guidelines exposures of 10CFR 100.

3.2.4 Applicability of the Quality Assurance Program to Other Safety-Related/Non Safety-Related Items

The following items are considered safety-related with some qualifying explanation.

1. Masonry Walls

   Masonry walls or "block" walls are treated as safety-related if,

   a. They provide support to any safety-related components or piping, or
   b. Failure of the wall could potentially threaten other safety-related structures, systems, or components.
2. Biological shielding within containment and auxiliary buildings and other radiation shielding. Installed biological shielding consists of reinforced concrete walls and masonry walls. These walls are considered safety-related to the extent that their structural integrity is required under seismic conditions.

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3.3 Wind and Tornado Loadings

All Category I 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.

Note:

This section of the FSAR contains information on the plant level bases and criteria for wind and tornado analysis and design. Additional information that may assist the reader is provided in the Plant Design Basis Specification for Tornado/Wind (CNS-1465.00-00-0008).

The Auxiliary Building Single Point Access door, door number S303A, is designed as a tornado pressure door. This door is normally held open, and will only be closed (by procedure) when there is a tornado warning.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The design wind velocity for all Category I structures is 95 mph, at 30 feet above the nominal ground elevation. According to ASCE Paper No. 3269, "Wind Forces on Structures" (Reference 1), this velocity is the fastest mile of wind with a recurrence interval of 100 years. This reference summarizes existing technical literature on wind velocities' distribution extending back to the early 1800's, and is the basis for the selected wind velocity.

Reference 2 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 I structures have a height/ width ratio of less than five. A gust factor of 1.10 is used for determining wind loads on all Category I structures except the Reactor Building. The wind load on the Reactor Building is determined as a function of the maximum height of the structure using the most severe coastal area criteria of Reference 1. Therefore, the effective velocity pressures used for design of Category I structures meet or exceed the values given in Table 5, Exposure C, of Reference 2. It is also concluded that wind forces on Category I structures are not a controlling load condition in the design.

The vertical distribution of velocity is discussed in Section 3.3.1.2.

3.3.1.2 Determination of Applied Forces

Reference 1 assembles existing information on the factors that determine applied wind loads on structures. Rectangular structures are designed for a wind distribution as defined in Reference 1. The wind pressure distribution, for the Auxiliary Building, Spent Fuel Pool Buildings, New Fuel Storage Buildings, and Doghouse Structures are shown in Figure 3-2. The wind pressure distributions for the Reactor Buildings above grade, for both the vertical and horizontal profiles are shown in Figure 3-1.

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

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

where: Gf = Gust factor as defined in Section 3.3.1.1 (not applicable to Reactor Building)

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 and Figure 3-2.

f = The constant obtained from Reference 1 for determining the dynamic pressure of the free stream.

v = The wind design velocity as defined in Section 3.3.1.1.

The Reactor Building design pressure distribution (which is proportional to the distribution of Cpe shown in Figure 3-1) is represented by a Fourier Series. Individual harmonics are analyzed by Kalnin's Computer Program as defined in Section 3.8.1.4, and are combined to produce the stress resultants of the total series.

The analysis of building parts and portions is not applicable to the design of Category I structures. The procedures delineated in Reference 4 for determining pressure coefficients for the Reactor Building gives results consistent with those derived using ASCE paper No. 3269 (Reference 1). ASCE paper No. 3269 (Part IV, Enclosed Structure Average Pressure Coefficients for Rounded Roofs) gives average values pressure distributions for segments of circular arches the center half and the windward and leeward quarters. Where, based on Reference 1, Figure 3-1 utilizes smaller segments resulting in slight variations in coefficient values.

3.3.2 Tornado Loadings

All Category I structures, except those structures not exposed to wind, are designed for tornado loads.

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 (Tangental) wind speed is 300 mph.
2. Translational speed of tornado is 60 mph.
3. Radius of maximum rotational speed is 240 feet.
4. Tornado induced negative pressure differential is 3 psi, occurring in three seconds.

The spectrum and characteristics of tornado-generated missiles is covered in Section 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

Tornado wind loadings are calculated for the various structures in accordance with procedures described in Section 3.3.1.2, using the velocities given in Section 3.3.2.1. The loadings from impact of tornado generated missiles is derived in accordance with procedures described in Section 3.5.3.

The loading combinations investigated are listed in Table 3-32.

3.3.2.3 Effect of Failure of Structures or Components 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 (East-West direction) due to tornado loading.

The design tornado has a peak rotational velocity of 290 mph at a radius of 150 feet and a translational velocity of 70 mph. (This is consistent with the tornado depicted in NRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants".)

Due to the size of the building in relation to the radius of the maximum tornado wind, an average design velocity over the areas of the building is calculated utilizing the tornado velocity distributions shown in Figure 3-3. The design velocity increases linearly from 70 mph at the center to 360 mph at a radius of 150 feet. The design velocity decreases as the radius increases above 150 feet according to the relationship shown in Figure 3-4.

Both possibilities of the siding being blown off or staying secured were investigated.

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

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 on Fragments Striking Structural Elements," Holmes and Narver, Inc. Anaheim , California, Revised November 1973.

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3.4 Water Level (Flood) Design

Note:

This section of the FSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document (DBD) for this system/structure, (CNS – 1465.00-00-0011).

3.4.1 Flood Protection

The plant yard elevation is 593.5 ft msl. This is above the calculated maximum static water elevation for the site of 592.2 ft. When combined with wave runup resulting from a 40 mph wind, the maximum water surface elevation becomes 593.7 ft msl. The minimum external access elevation for the auxiliary, turbine, and service buildings is 594.0 ft msl. The plant site is provided with a surface water drainage system that protects all safety-related facilities from flooding during a local PMP (See Section 2.4.2.3).

The crest of the NSWP dam is higher than the maximum flood elevations. The side slopes are protected from wave action by riprap.

A permanent Category I groundwater drainage system is installed to maintain a normal groundwater level near the base of the foundation mat and basement walls, thus eliminating the hydrostatic forces on the Auxiliary and Reactor Buildings (See Section 2.4.13.5 and Section 9.5.11). The testing and inspection on the ground drainage system are performed in accordance with SLC 16.7-8.

As a result of these measures, there are no safety related systems or components that are either completely or partially submerged by the design basis flood.

3.4.1.1 Regulatory Guides

Regulatory Guide 1.59, Design Basis Floods for Nuclear Power Plants: Refer to Section 1.7.

Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants: Refer to Section 1.7.

3.4.2 Analysis Procedures

The NSW and SNSW intake and discharge structures are both fully submerged at all times and are not subject to the effects of flooding.

The NSW openings and SNSW pump structure are at elevation 599.5 which is above the flood levels stated in Section 2.4.2.2, and therefore is not subject to the effects of flooding. The buoyant effect on this structure has been checked to determine the safety factor against floatation using two criteria:

1. Weight of the structure including all equipment with one sump filled and the other sump empty, versus the maximum flood level of El. 596.8. This produced an FS = 1.63.
2. Weight of the structure including all equipment with both sumps empty, versus the water table at the assumed level of El. 576.0. This produced an FS = 2.15.

The hydrostatic level on the SNSW Pond Dam is considered to be at an elevation of 583.5, which is probable maximum flood level. This elevation is higher than the possible ground-water level of 574.0, which is the normal SNSW pond level. The hydrostatic level on the NSW and

SNSW Electric Conduit Manhole and the pipe trench to the Reactor Make-Up and Refueling Water Storage Tanks is assumed to be at yard elevation of 593.5. Groundwater pressure was applied as static pressure.

The bouyant effect on the NSW Electric Conduit Manholes, using the worst case of lightest conduit manhole of greatest depth, and the water table at the yard elevation (El. 593.5) produced an FS = 1.53.

The buoyant effect on the pipe trench is considered as having no effect because the trench is shallow and located above normal groundwater level.

No other Category I structures are subject to hydrostatic forces unless identified in Section 3.8.4 or Section 3.8.1.

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3.5 Missile Protection

> **Note:**
>
> This section of the FSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document (DBD) for this system/structure.

3.5.1 Missile Selection and Description

The systems located both inside and outside 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.

3.5.1.1 Internally Generated Missiles (Outside Containment)

The structures that are to be protected against damage from internally generated missiles outside containment are listed in Table 3-7. The essential safety-related equipment within these structures are those whose loss could lead to conditions in excess of the guidelines specified in 10CFR 100. Items in this category are those in which a single strike by a potential missile could result in a loss of the capacity to function in the manner needed to meet these guidelines.

In the unlikely event that a missile should be generated, separation of redundant safety-related systems should prevent a compromise of plant safety.

The two basic modes of missile generation within these structures are as follows:

1. Failure of pressurized components.
2. Failure of rotating machinery.

These two failure modes are restricted to safety-related ASME Section III components and are discussed individually in Section 3.5.1.1.1 and Section 3.5.1.1.2.

A review of non-safety related equipment and valves in the Reactor Building, Doghouse, and Auxiliary Building was made to identify potential sources of missiles. It is concluded that there is very low probability that missiles would be generated from non-safety related rotating machinery and pressurized equipment and valves. Much of the subject equipment and valves are Duke Class F, designed to the same seismic specifications as seismic Category I equipment. Class F valves are designed to ANSI B31.1. In addition non-seismic piping and equipment interactions were reviewed to determine and resolve any adverse effects non-seismic piping, valves and equipment could have on safety related equipment. From their reviews it was confirmed that non-safety-related components are of insufficient size and energy to do significant damage.

3.5.1.1.1 Failure of Pressurized Components

Catastrophic failure of the pressure boundary of ASME Section III components 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 conservative design, material characteristics, inspections, quality control during fabrication, erection and operation of the particular component.

These valves, pumps, heat exchangers and piping are all designed, fabricated, inspected and tested in strict accordance with Section III of the ASME Boiler and Pressure Vessel Code for

Nuclear Power Plant Components. The quality of these items is monitored and assured by the quality assurance organizations of both the manufacturer and Duke Power Company.

In order to prevent overstressing pressure boundary bolting, proper control is exercised to prevent over-torquing the bolts during tightening.

All pressurized equipment and sections of piping that from time to time may become isolated under pressure are provided with ASME Section III Code acceptable pressure relief valves. These valves will be preset to ensure that no pressure buildup in equipment or piping sections will exceed the design limits of the materials involved.

The design pressure and temperature for pressurized equipment and piping are conservatively determined in order to allow for the maximum achievable pressure and temperature within their system. All equipment is hydrostatically tested in accordance with Section III of the ASME Boiler and Pressure Vessel Code. All completed systems are then subjected to a final hydrostatic test before operation.

In order to prevent ejection of valve stems, all high pressure valves have stems with back seats. Additional interference is encountered from air, gear and motor operators. For these reasons, valve stems are not considered to be credible sources of missiles.

3.5.1.1.2 Failure of Rotating Machinery

Pumps located outside containment have been evaluated for missiles associated with overspeed failure. The maximum no-load speed of electric motor driven pumps is equivalent to the synchronous speed of their associated motors. Consequently, no pipe break or single failure in the suction line will increase the pump speed over that of the no-load condition. The only rotating components which could have a credible overspeed condition are the auxiliary feedwater pump and turbine and the diesel generators.

The manufacturer of the auxiliary feedwater turbine has tested the solid wheel turbine under overspeed conditions and no missiles were generated. The pump itself may develop missiles under overspeed conditions but its potential for damage is small because of the small size of any missiles postulated. The room containing the pump is designed in such a manner as to minimize the potential for damage caused by postulated pump missiles. In the extremely unlikely event that the governor-control valve speed control should fail, a highly reliable mechanical overspeed trip will activate the stop valves thereby preventing destructive turbine overspeed. This mechanical overspeed trip is provided with manual and electrical backups located in both the control room and at the turbine. Regular in-service inspection of the turbine, governor-control valve system and overspeed trip components assures the proper operation of the auxiliary feedwater pump turbine. Finally, non-destructive examination of pump impellers and turbine wheels serves to detect material flaws. By repairing or rejecting flawed materials, the integrity of the pumps and turbines is assured.

Four emergency diesel generators, which are required to supply emergency power to certain engineered safety features, are located inside two separate structures, the Diesel Generator Buildings. Interior walls of reinforced concrete separate these generators.

There is a mechanical governor on the diesel engine of each diesel-generator unit which is designed to assume control of the engine when there is a tendency to overspeed. In addition, the diesel generators have an overspeed trip which cuts off fuel to the diesel engine upon an overspeed condition. Consequently, no missiles are postulated for overspeed conditions of the generators. Regular in-service inspection of the speed control and overspeed trip components serves to assure the integrity of the diesel generators.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The essential safety-related equipment within the containment as defined in Section 3.5.1.1 have been examined to identify and classify potential missiles. As noted before, in the unlikely event that a missile should be generated, separation of redundant safety-related systems should prevent a compromise of plant safety.

Components which, nevertheless, are considered to have a potential for missile generation are the following:

1. Reactor coolant pump flywheel
2. Control rod drive mechanism shafts and/or housings
3. Valves
4. Instrument wells and thimbles
5. Pressurizer heaters

These components are discussed in more detail below.

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casing 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 conservative design, material characteristics, inspections, quality control during fabrication, erection and operation of the particular component.

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

3.5.1.2.1 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 machine 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. Its rotational speed is therefore 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. The most adverse operating condition of the pump motor flywheel is visualized to be the loss of outside load situation. During a loss of outside load condition at the plant, the turbine generator will

overspeed and the reactor coolant pump will tend to follow. Analyses of the turbine control system with the utilization of redundant inlet valves (turbine stop and governor valves) and redundant re-heat valves (re-heat stop and intercept valves), show that the maximum turbine overspeed under such conditions will not exceed 120 percent and would not persist for more than 30 seconds.

Bursting speed of the flywheel can be calculated on the basis of Griffith-Irwin's results (Reference 1) to be 3900 rpm; more than three times the operating speed.

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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.

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 ten years, will be more than adequate as part of a plant surveillance program. Evolution of the inspection program has included a move to an in-place volumetic UT exam covering the volume from the bore to a circle at one-half of the outer radius OR the surface inspection of the removed flywheel. This was supported with WCAP-14535A. Subsequently the inspection period was further relaxed to every 20 years as supported with WCAP-15666.

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 bearing and in the shaft due to rotational movements of the pump about a horizontal axis during these seismic disturbances. For the SSE, the pump shall maintain its pressure boundary integrity and flow coastdown capability.

3.5.1.2.2 Control Rod Drive Mechanism

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

1. Control rod drive mechanisms are shop hydrotested at 4100 ± 75 psi.
2. Control rod drive mechanism housing are individually hydrotested to 3107 psi after they are installed on the reactor vessel to the head adapters, and checked again during the hydrotest of the completed reactor coolant system.
3. Control rod drive mechanism housing are made of Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

However, it is postulated that the top plug on the control rod drive mechanism will become loose and it will be forced upward by the water jet. The following sequence of events is assumed: The drive shaft and control rod cluster are forced 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 12 feet of travel, the rod cluster control spider hits the underside of the upper support plate. Upon impact the flexure arms in the coupling joining the drive shaft and control cluster fracture, completely freeing the drive shaft from the control rod cluster. The control cluster would be completely stopped by the upper support plate. The drive shaft would continue to be accelerated upward to hit the missile shield provided.

The control rod drive mechanism missiles are summarized on Table 3-8. The velocity of the missiles is calculated by balancing the forces due to the water jet. No spreading of the water jet has been assumed.

3.5.1.2.3 Valves

As noted in Section 3.5.1.1.1, due to their ASME Section III design, inspection, testing, back seats and operators, valves are not considered to be credible sources of missiles.

3.5.1.2.4 Instrumentation Wells and Thimbles

The only credible source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is that represented by the temperature and pressure sensor assemblies. The resistance temperature sensor assemblies can be of two types ("with well" and "without well"). Two rupture locations have been postulated: one around the welding between the boss and the pipe wall and another at the welding (or thread) between the temperature element assembly and the boss for the "without well" element, and the weld (or thread) between the well and the boss of the "with well" element.

A temperature sensor 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 of a steel plate. In evaluating missile potential, it is assumed that this plate could break and the pipe plug on the external end of the hole could become a missile.

In addition, it is assumed that the welding between the instrumentation well and the pressurizer wall could fail and the well and sensor assembly could become a jet-propelled missile.

The missile characteristics of the piping temperature sensor assemblies are given on Table 3-9. A 10 degree expansion half angle water jet has been assumed. The missile characteristics of the piping pressure element assemblies are less severe than those of Table 3-9.

The missile characteristics of the reactor coolant pump temperature sensor, and the instrumentation well of the pressurizer are given on Table 3-10. A 10 degree expansion half angle water jet has been assumed.

3.5.1.2.5 Pressurizer Heaters

The heaters are installed underneath the bottom head of the pressurizer. The characteristics of this type of missile are noted in Table 3-10.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

There are two basic modes of failure which may result in the generation of turbine missiles:

1. Destructive overspeed failure which results in wheel failure by general ductile yielding due to a failure of the turbine overspeed protection control. Studies of hypothetical runaway conditions indicate that destructive overspeed failures will occur at approximately 175 percent of rated speed.
2. Design overspeed failure which results in wheel failure due to brittle fracture at or near normal operating speeds. Design overspeed failures require the combination of both a severe stress concentration and a brittle material state. Stress concentrations can be produced by internal material discontinuities such as cracks, nonmetallic inclusions or alloy

segregates. The susceptibility to brittle fracture in the presence of such stress concentrations is measured by the temperature at which failure behavior changes from ductile to brittle.

Missiles from turbine failures can be divided into two groups:

1. High Trajectory Missiles which are ejected upward through the turbine casing and may cause damage if the falling missile strikes an essential structure.
2. Low Trajectory Missiles which are ejected from the turbine casing directly towards an essential structure.

The essential safety-related structures at the plant are those whose damage could lead to conditions in excess of the guidelines specified in 10CFR 100. Structures in this category are those in which a single strike by a potential turbine missile could result in a loss of equipment capability to function in the manner needed to meet these guidelines. At the Catawba Nuclear Station, these structures are noted in Table 3-7.

The above safety-related structures are not designed to be turbine missile barriers. The minimum surface area and location of these structures, however, result in an extremely low probability that a turbine missile would strike and damage an essential system.

3.5.1.3.2 Turbine Placement and Plant Orientation

The locations of these essential safety-related structures as well as their relationship to the potential turbine missile sources are shown in Figure 3-5.

The majority of the structures are located either along or within close proximity to the longitudinal centerlines of the respective turbine complexes. This orientation greatly reduces the number of safety related structures located within 25 degrees of the plane of the outboard wheels. Evidence currently available indicates that structures outside this critical 25 degree zone are not endangered by high-energy low trajectory missiles. The following structures are the only safety related structures within this critical 25 degree zone:

1. Nuclear Service Water Pump Structure
2. Diesel Generator Buildings
3. Auxiliary Building Switchgear Rooms
4. Main Steam and Feedwater Piping Restraints

For high trajectory missiles, the potential strike area is found to depend on the initial energy (i.e., velocity) of the particular missile and on the angle of elevation at which the missile is released. It is found that if the angle of elevation is less than 85 degrees there is little likelihood that a missile would cause damage to critical plant structures. Missiles released at lower angles generally do not have the combination of high velocity (350 ft/sec or greater) and short range (1000 ft or less) required to damage critical plant structures. This elevation limit creates a critical 5° x 10° window through which the missiles must pass. All safety related structures are assumed to be in the range of these missiles.

3.5.1.3.3 Missile Identification and Characteristics

Based on hypothetical turbine missile data provided by General Electric, 64 different combinations of turbine stage groups, wheel fragment groups and energy groups are postulated. General Electric recommends groupings as noted below:

I=Stage Groups	J=Fragment Group	K=Energy Group
1. Stage 7, 120% speed	1. 120° sector, 2 fragments/ group	1. Low
2. Stages 1-3, 180% speed	2. 60° sector, 1 fragment/group	2. Mid-low
3. Stages 4-6, 180% speed	3. Small, 3 fragments/group	3. Mid-high
4. Stage 7, 180% speed	4. Extra small, 10 fragments/ group	4. High

The General Electric data used to derive these groups is noted in Table 3-11. When considering brittle fracture at low speed, General Electric has determined that the seventh stage wheel is several orders of magnitude more likely to fail than the inner stage wheels by virtue of its large mass. For this reason, only the seventh stage is considered as a possible source of missiles due to a design overspeed failure. It is assumed that only one wheel of one low pressure turbine will fail producing 16 missile fragments. The assumed fragment sizes are based on results of destructive tests performed by General Electric on similar turbine wheels. The four different energy levels are spread over the possible energy range for the stage group being considered to account for the resistance to missile ejection provided by the turbine shell. This resistance is assumed to range from zero to complete containment of the missile.

3.5.1.3.4 Turbine Missile Hazard Evaluation

The turbine missile protection criterion utilized in the design of the Catawba Nuclear Station is that the probability of unacceptable damage should not be significant. An event having a probability of causing unacceptable damage of the order of about E-7 per reactor year at the plant is not considered significant. For the two-unit Catawba Nuclear Station, an event having a probability of occurrence of the order of 2 E-7 is sufficient to fulfill this criterion. The probabilities presented in this evaluation are considered for the 40 year lifetime of the plant.

To arrive at a credible number to represent the probability of such an accident actually occurring an analysis can be performed utilizing a memo report from General Electric (Reference 2) and several conservative assumptions.

The total probability of damage resulting from a particular turbine missile, P4, can be thought of as the product of P1, P2 and P3, where:

P1 = Probability of generation and ejection of a high energy missile

P2 = probability that a missile, having been ejected, strikes a critical target

P3 = probability that a missile, having struck a critical target, results in unacceptable damage

This total probability is therefore dependent on the assumptions made for the missiles produced (i.e., number, size, and energy of missiles, etc.). High trajectory and low trajectory missiles have been evaluated separately in Section 3.5.1.3.4.1 and Section 3.5.1.3.4.2.

The analysis presented in this section and associated Tables (in particular Tables 3-13 and 3-14) do not accurately reflect the current Licensing Basis. More recent turbine missile probabilities provided by the turbine manufacturer, GE, have been used to optimize turbine stop/control/intercept valve surveillance intervals specified in Selected Licensee Commtment (SLC) 16.7-5. This analysis is retained here for illustrative purposes to show how the plant was originally evaluated with respect to turbine missile hazards. The acceptance criteria discussed in the Safety Evaluation Report (SER, section 3.5.1.3), and above are still satisfied with this

optimized valve testing. Further discussion on this subject is documented in the 10CFR50.59 evaluation for the SLC change (reference 12).

Calculations identified in reference 11 document the existing current turbine missile probabilities based on the most recent rotor inspection data. Total unit turbine missile probabilities will be updated as they become available. These calculations will document and determine whether future changes in turbine Stop/Control/Intercept valve testing frequencies (surveillance intervals) need to be made. For example, a crack could be identified in the future of sufficient size, to require more frequent valve testing, rotor inspections, turbine pre-warming, or a combination of the three, to provide an acceptable level of confidence that a turbine missile will not be generated.

The current approach assumes that P1 (probability of turbine missile being generated) is the only variable, as rotor inspection data is identified through future rotor inspections. Implicit to this assumption is the position that P2xP3 (combined strike and damage probability) is constant. That is, no significant structural modifications have been made to the critical targets identified in FSAR section 3.5.1.3.2 (or no equally important structures, systems or components (SSCs) have been added to the site) to change the quantity P2xP3. Thus, monitoring P1 is sufficient to assure the total plant damage probability limit of 1E-7 per reactor year is preserved.

3.5.1.3.4.1 High Trajectory Missiles

In the actual calculations, a value for P4 is determined for each of the 64 cases listed in Section 3.5.1.3.3; and the total probability of damage is assumed to be the sum of the P4's. For the individual calculations, the following expression is used for P4.

$$P4 = P1AxP2xP3xA$$

P1A is a function of the stage and fragment groups above, and is the General Electric value for P1 modified as follows:

$$P1A(I, J) = \frac{P1(I)xNHxNF(J)}{HxNR}$$

where

P1(I)	=	GE value for probability of turbine rotor failure, per stage group for a three low pressure turbine unit operating for 40 years.
NH	=	Number of low pressure turbine hoods considered in the calculation (NH=1)
NF(J)	=	Number of fragments in the particular fragment group being considered
H	=	Number of hoods in turbine unit (H=3)
NR	=	Number of energy ranges being considered (NR=4)

General Electric has determined that the probability of a missile being generated by brittle fracture at low speed (127% speed) is 3.46E-7 and that the probability of a missile being generated by ductile fracture at runaway speed (180% rated speed) is 2.0E-7. These values are for a 1800 RPM tandem compound six flow machine with 43 inch last stage buckets, considered for a 40 year lifetime. (Reference 3) The values for percent overspeed are based on the characteristics of the turbine and control system. The highest transient speed the turbine can reach is 127% rated speed assuming proper operation of either of the two overspeed trip systems, failure of all intermediate stop and intercept valves, and successful operation of either all main stop or all main control valves. The speed at which General Electric calculates loss of

the last stage buckets is 180% overpseed. This value of 180% assumes one or more main stop valves and one or more main control valves fails to close. (Reference 4)

To determine the probabilities given above, General Electric considers three operating modes - cold startup from 0 percent to 180 percent speed, a test of the emergency trip mechanism from 110 to 180 percent speed, and a sudden loss of load from 110 to 180 percent speed. The lifetime probability of operation in each of these modes is considered. For speeds above 120 percent, probabilities of failure of control components are calculated into the lifetime probability of operation. These probabilities are computed by using failure rates for electronic components, sticking rates for valves, failure rates for the mechanical overspeed trip, and failure rates for the hydraulic system. Only components or systems similar to General Electric's present day design are considered. For all speeds the probability of fracture of the wheels is considered. Results from destructive tests with failures by both brittle and ductile fractures are combined with non-destructive test results which relate the test specimens to actual turbine wheels. This approach results in a single relationship for the conditional failure of a wheel, by either brittle or ductile fracture.

In order to determine P1(I), the above failure probabilities have to be distributed among the four stage groups. Due to the higher probability of failure of the seventh stage wheel, the entire low speed failure probability of 3.46 E-7 is assigned to the first stage group. When considering ductile failure at high speed, the probability of failure is about the same for each stage; so the value of 2.0E-7 for ductile failure is divided equally between the seven stages. The probability of ductile failure of any particular stage is therefore 2.86E-8. Subsequently the values of P1(I) for the second, third and fourth stage groups are 8.58E-8, 8.58E-8 and 2.86E-8 respectively.

P2 can basically be described as strike probability with some important qualifications that will be discussed later. The expression found for P2 relates the 5° by 10° critical window discussed in Section 3.5.1.3.2 to the total possible angles of release for turbine fragments. These possible angles are a full 360° around the circumference of the rotor, and within ±5° of the plane of the wheels for turbine stages 1-6 (inner stages). This angle widens to ±25° for the seventh stage wheels. The expression for P2 also considers the total possible impact area for missiles being ejected through the critical window. P2 for a particular high trajectory missile is determined as the follows:

$P2(I,J,K) = 47.7/ \ [Vo(I,J,K,)]^4$ for last stage wheels (I=1,4)

$P2(I,J,K) = 238.5/ \ [Vo(I,J,K,)]^4$ for inner stage wheels (I=2,3)

Vo(I,J,K) is the initial velocity of the missile along its trajectory. These expressions give P2(I,J,K) for a unit area, neglecting atmospheric drag.

Table 3-12 tabulates the strike probabilities for the critical structures noted in Table 3-7. In order to derive these probabilities, it is necessary to multiply the above expression for P2 (I,J,K) by both the structure area, noted in Table 3-7, and the number of fragments ejected, discussed below.

In keeping with our earlier assumption, only one wheel of one low pressure turbine will fail producing 16 fragments or missiles. A specific strike probability for each fragment group is determined by finding average values of P2 (I,J) for the four different valves of K (or four energy levels). This averaging results in 16 values of P2 (I,J) from the 64 different P2 (I,J,K) generated by our analysis. These average P2 (I,J) are then multiplied by the number of fragments in the respective fragment groups N (J), and the products summed for each stage group. The resulting value of P2 (I) for stage group 1 gives the unit area strike probability for the design overspeed (120% speed) failure mode. By averaging the respective values of P2 (I) for stage

groups 2 through 4 (turbine stages 1 through 7), one obtains the unit area strike probability for the destructive overspeed (180% speed) failure mode.

In calculating the total probability of damage, P4, the number of fragments and the area are not included in the strike probability, P2. The number of fragments, N(J), is already accounted for in the generation probability PIA. The total unit area damage probability (PIA X P2 X P3) is multiplied by the structure area.

It must be kept in mind when using Table 3-9 that the values for strike probability are valid only for the particular stage, fragment, and energy groups assumed in the basic analysis. If another analysis were performed in which J (fragment group) is assumed to be one thousand, the strike probability will increase significantly. This increase will be offset, however, by a decrease in penetration probability; since an assumption of a large number of missiles necessarily involves an assumption of smaller missiles. Since this analysis assumes that any missile ejected through the 5° by 10° critical window has an equal chance of hitting any unit area within its range, strike probabilities are assumed equal for missiles from all six low pressure hoods. All critical plant structures are assumed to be located within the missile distribution area defined by the critical window.

P3(I,J,K) is the probability that a missile that strikes a building will actually cause damage. Following the recommendation of Spencer H. Bush (Reference 5) and others, the Duke analysis uses the Petry formula, modified for finite thickness, to calculate the critical thickness of concrete required to prevent penetration of the missile. This formula relates velocity, penetration coefficient, missile weight and size to determine the critical thickness. The effective thickness of the concrete slab is compared to the critical thickness. For flat surfaces, such as the Auxiliary Building roof, the effective thickness is equal to the slab thickness; and P3 (I,J,K) will be either one or zero.

For curved surfaces such as the Reactor Building dome, the effective thickness will be a function of the horizontal distance from the center of the dome to the point of impact of the missile. For these curved surfaces, the concrete thickness and critical thickness are used to determine a "penetrable area" for each of the 64 cases considered. P3 is then the ratio of the penetrable area to the total building area.

P3 (I,J,K) is obviously dependent on the compressive strength of the concrete (which determines the concrete coefficient in the Petry formula) and the concrete slab thickness. P3(I,J,K) is also dependent on the size and energy of the particular missile being considered. The critical thickness will be much greater for large, high energy fragments than for smaller fragments. Such high energy fragments, incidentally, will have a low value for P2(I,J,K); since the same missile characteristics which tend to make P2 large (a great many low-velocity missiles) cause P3 to approach zero. Table 3-13 lists values of the critical thickness and P3(I,J,K) for concrete of 3000 lb. compressive strength (which corresponds to Catawba design). Table 3-13 also lists final values of P4 (I,J,K) for each missile as well as the total damage probability for each structure. It should be noted that these probabilities are based on the failure of one one stage of one low pressure turbine hood. Table 3-14 lists values for the total damage probabilities for the entire plant. These values are corrected to account for the higher missile generation probability associated with six low pressure turbine hoods as well as multiple structures within the plant.

The other term in the expression for P4(I,J,K) is the building area (A) under consideration.

3.5.1.3.4.2 Low Trajectory Missiles

As noted in Section 3.5.1.3.2, only structures located within the critical 25° zone are considered to be targets for low trajectory missiles. The few structures located within this zone are excluded from probability analysis for the following reasons:

1. The small surface area (3700 sq ft) and large range (2150 ft) of the Nuclear Service Water Pump Structure would result in an extremely low strike probability. The protected redundant systems within the pump structure further reduce the possibility of unacceptable damage.
2. The roofs of both the Diesel Generator Buildings and the Auxiliary Building Switchgear Rooms are located at ground level. As these structures are located below the turbine elevation, any turbine missile would have to pass through equipment, piping, walls, concrete floors and structural steel before striking these structures. In addition, the strike angle would be extremely shallow which would further reduce the possibility of damage. Finally, redundant systems within these structures would assure the safety of the plant.
3. The Main Steam and Feedwater Piping Restraints are also located below the turbine elevation, and therefore, are protected as noted above. As noted in Figure 3-5, only one set of restraints are located within the critical zone per unit. For this reason, should a missile destroy a restraint and break the attached piping, the remaining set of piping and restraints should allow for the safe shutdown of the plant. Finally, due to the small surface area of the restraints and their position with respect to the turbine, the probability that a missile, which does penetrate the Turbine Building, striking a restraint will be extremely small.

3.5.1.3.5 Conclusion

The probabilities that a turbine wheel will fail and the resulting missiles strike and damage a safety-related structure can be seen to be significantly small. Furthermore, it should be noted that damage to a safety-related structure does not necessarily result in significant damage to essential safety-related systems therein.

In order to assure the integrity of the turbine wheels, and therefore, reduce the possibility of failure, Duke Power incorporates the following into the design, construction and operation of the Catawba Nuclear Station:

1. Turbine control systems to prevent overspeed of the turbine wheels with resulting ductile fracture failure. These highly reliable, redundant overspeed control systems assure that turbine overspeed is controlled within 120 percent of rated speed. A turbine generator operating at full load which fails to trip, for any reason, at the first emergency setting of 110 percent of rated speed will be tripped by the redundant back-up overspeed trip system at 111.5 percent of rated speed. Thus, two separate overspeed control systems initiate a trip of the unit at 110 percent and 111.5 percent of rated speed. The maximum turbine speed following trip by either control system should be less than 120 percent of rated speed. The turbine speed range in which reliable, redundant overspeed controls will be initiated is quite conservative compared to the range at which probable wheel failure will occur. Refer to Section 10.2.2 for a more detailed discussion of turbine control.
2. Quality assurance of turbine wheel material to detect flaws which could lead to brittle fracture failure.

   The program of quality assurance applied to turbine manufacture includes ultrasonic examination of each turbine wheel to assure that no wheel possesses any questionable indications.

Following this test, all machined surfaces will be subjected to a magnetic particle test and each fully bucketed turbine wheel will be spin tested. Refer to Section 10.2.3 for a more detailed discussion of turbine wheel quality assurance.

3. In-service inspection of turbine wheels and speed control valves at regular intervals to detect malfunctioning components before they fail. Refer to Section 10.2.3.6 for a more detailed discussion of in-service inspection procedures.

3.5.1.4 Missiles Generated by Natural Phenomena

The only postulated missiles generated by natural phenomena are tornado generated missiles. These tornado generated missiles are per the NRC Standard Review Plan 3.5.1.4 (Revision 1) and Regulatory Guide 1.76.

Table 3-15 lists the dimensions, mass, energy, and design velocity of all postulated tornado generated missiles.

The structures used for tornado generated missile protection are identified in Table 3-1. All designated structures that are designed for tornado generated missile protection possess a concrete strength of 4000 psi or 5000 psi (based on a 28-day curing time). The minimum thickness of a tornado missile protection barrier is 24 in.

The railroad freight door into the new fuel storage building was not designed for tornado missile impact. There are no safety related systems or components that can be impacted if the missile were to penetrate the freight door.

3.5.1.5 Missiles Generated by Events Near the Site

There are no missiles postulated due to events near the site. A further discussion of near site events is contained in Section 2.2.

3.5.1.6 Aircraft Hazards

There are no missiles postulated due to aircraft accidents. For a detailed discussion of aircraft hazards see Section 2.2.

3.5.2 Systems To Be Protected

All plant structures, systems and components whose failure can lead to offsite radiological consequences or which are required to shutdown the reactor and maintain it in a safe condition, assuming an additional single failure, are listed in Section 3.2.

The following is a list of the safety-related components that are exposed to tornado generated missiles and the protection provided each. The barrier design procedures are provided in Section 3.5.3.

1. The Refueling Water Storage Tank is protected by a missile-proof barrier wall capable of containing a sufficient quantity of refueling water in the event the storage tank is ruptured.
2. The HVAC System air intakes located in the Fuel Building, and the steam exhausts in the doghouse, are protected with "gull wings." The steam exhausts in the Upper Head Injection Building have blow-off shafts and the diesel exhausts in the Diesel Generator Building have concrete protective shafts which, like the "gull wings," absorb the energy of tornado missiles and prohibit them from entering the building and possibly damaging safety-related equipment.

3. The Ground Water Drainage Sump is protected by a three foot thick barrier wall and a two foot thick roof above.
4. The Electrical Conduit Manholes have one and one-half foot thick hatch covers designed to withstand the impact of tornado missiles. The Electrical Conduit running between manholes is covered with a one foot thick concrete slab when it does not have five feet or more of ground cover.
5. The pipe trench running from the Reactor Building to the Refueling Water Storage Tank is protected by an eighteen inch reinforced concrete cover or eight inch reinforced concrete cover with five feet or more of earth cover.
6. The three doors located at elevation 594+0 of the Auxiliary Building are constructed of two inch thick plate steel, and are designed to withstand the impact of tornado missiles.
7. All other doors not designed for tornado missile impact are located such that missiles entering through their openings could not damage safety-related equipment.
8. The equipment hatch in the Reactor Building is constructed of three foot thick concrete.
9. The opening for the vent in the Auxiliary Building is located adjacent to the Reactor Building and the Doghouse Structure. This opening is shielded by these adjacent structures from the possibility of missile impact.

3.5.3 Barrier Design Procedures

3.5.3.1 Local Damage Prediction

The Catawba Nuclear Station Category I structures are designed to resist the effects of tornado generated missiles in combination with other loadings as outlined in Section 3.8.1. Table 3-15 lists the postulated tornado generated missiles. All missiles are per the NRC Standard Review Plan 3.5.1.4 Rev. 1 and Regulatory Guide 1.76.

Penetration depths are calculated using the modified NDRC formula and the modified Petry formula (References 6 and 7). Table 3-16 lists the calculated penetration depths and the minimum barrier thicknesses to preclude perforation and scabbing hence eliminating secondary missiles. The barrier thicknesses for all category I structures are such that they preclude any perforation and/or scabbing from the impact of any of the postulated tornado generated missiles. Minimum barrier thickness is three times the postulated missiles calculated depths of penetrations (See Table 3-16).

All barriers are reinforced concrete, except for metal doors designed for missile impact. No other steel or composite barriers are used.

3.5.3.2 Overall Design Procedures

Applied missile force time histories are calculated based on the procedures outlined in Reference 7 and Reference 8.

Portions of the Reactor Building shell and dome are modeled using a space frame finite elements model subject to the tornado missile loads. The results of these analyses in terms of resulting forces, moments and shears are combined with other loadings on the structure as specified in Table 3-32.

After demonstrating that the postulated missiles would not penetrate the barrier, an equivalent static load, concentrated at the impact area, was determined from which the structural

response of the barrier, in conjunction with other design loads, was evaluated using conventional design methods. The response of a structure to missile impact depends largely on the location of impact (midspan of a slab, or near the support), on the dynamic properties of the target and missile, and on the kinetic energy of the missile. The collision is assumed to be plastic, that is, all of the kinetic energy of the missile is absorbed into structural strain energy. However, energy losses due to missile deformation and local penetration are considered when applicable.

For missiles 1 through 5, the impact force is calculated based on the following formula from Reference 7:

$$F_i = \frac{WV^2}{gx}$$

where

F_i = force of impact (lb)

W = weight of fragment(lb)

V = velocity of impact(ft/sec)

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

x = penetration (ft)

For missile 6, the derivation of a force-time history is calculated based on the following formula from Reference 8:

$$F(t) = 0.625 V_s W_m \sin 2Dk (0 \le t \le 0.0785 \sec)$$

$$F(t) = 0 (t > 0.0785 \sec)$$

where

t = time from the instant of initial contact (sec)

F(t) = time-dependent force on target (lb)

V_s = striking velocity of the automobile (ft/sec)

W_m = weight of automobile (lb)

The ductility ratio (μ) is defined at the ratio of the maximum acceptable displacement ($\Delta\mu$) at failure to the corresponding displacement ($\Delta\mu$) at initial yield (Reference 9). The ductility ratio ranges from 1.0 for a completely brittle structure, to 100.0 for a very ductile structure. For moderately brittle structures, μ=3 to 5; for moderately ductile structures, μ=10 to 30 (Reference 10). The ductility ratio for reinforced concrete structures depends on the quantity of reinforcement, on the relative quantities of tensile and compressive steel, the compressive strength of concrete the yield stress of steel reinforcement, and the structural behavior of the member under consideration (i.e., whether it is a beam, column, or shear wall). Structures designed in accordance with ACI-318-71 generally have adequate ductility. In critical areas, however, ductility of the structure may be further improved to assure adequate rotational capacity of the members and redistribution of moments in the structure by use of one of the following:

1. Compressive reinforcement
2. Closed stirrups to confine the concrete

3. Under-reinforced sections
4. High strength concrete
5. Reinforcement bars with lower yield stress

The ductility ratio used in the analysis and design of concrete structures does not exceed 10 unless it can be shown by elasto-plactic analysis, considering both local and overall collapse of the structure, that various elements of the structure have adequate strength and rotational capacity required for the redistribution of moments over the entire structure.

The ductility ratio used in the analysis of concrete structures was 1.0 for shear and 10.0 for bending moment.

3.5.4 References

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3.6 Protection Against Dynamic Effects Associated With The Postulated Rupture of Piping

Note:

This section of the FSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document (DBD) for this system/structure.

General Design Criterion 4 of Appendix A to 10CFR 50 required that structures, systems, and components important to safety be protected from the dynamic effects of pipe failure. This section describes the design bases and design measures to ensure that the containment vessel and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary, have been adequately protected against the effects of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping.

Criteria presented herein regarding break size, shape, orientation, and location are in accordance with the guidelines established by NRC Regulatory Guide 1.46, and include considerations which are further clarified in NRC Branch Technical Position MEB 3-1 and APCSB 3-1 where appropriate. These criteria are intended to be conservative and allow a high margin of safety. For those pipe failures where portions of these criteria lead to unacceptable consequences, further analyses will be performed. However, any alternative criteria will be adequately justified and fully documented.

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6.1.1 Design Bases

3.6.1.1.1 Reactor Coolant System

The Reactor Coolant System, as used in Section 3.6 of the Safety Analysis Report, is limited to the main coolant loop piping and all branch connection nozzles out to the first butt weld. Dynamic effects are only considered for pipe breaks postulated at branch connections. 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 connections. Consequently, pipe whip and jet impingement effects of the postulated pipe break at this location will not result in unacceptable consequences to essential components. This restraint configuration, along with the particular arrangement of the Reactor Coolant System and building structures, mitigates the effects of the jet from the given break such that no unacceptable consequences to essential components are experienced.

The application of criteria for protection against the effects of postulated breaks at the branch connections 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 bases for postulated breaks in the Reactor Coolant System are discussed in Section 3.6.2.1.

3.6.1.1.2 All Other Mechanical Piping Systems

This section discusses all piping systems excluding the Reactor Coolant System as described in Section 3.6.1.1.1 and is in accordance with NRC Branch Technical Position APCSB 3-1 and Regulatory Guide 1.46 except as noted in Table 3-19.

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

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 one or both of the following are met:

1. Maximum operating temperature exceeds 200°F or
2. Maximum operating 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 in place of the 200°F criterion. Moderate-energy through-wall cracks as defined in Section 3.6.2.1.2.3 are assumed in these piping systems and the environmental effects of pressure, temperature, humidity, flooding, and wetting of equipment are all considered in the station analysis.

Systems are classified as moderate-energy if the total time that either of the above conditions 1) or 2) are met is less than either of the following:

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

Moderate-Energy Piping Systems are those systems, or portions of systems, that during normal plant conditions are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

1. Maximum operating temperature is 200°F or less (or less than the atmospheric boiling temperature for non-water systems), and
2. Maximum operating pressure is 275 psig or less.

Systems which do not contain mechanical pressurization equipment are excluded from moderate-energy classification (e.g., systems without pumps, pressurizing tanks, boilers, or those which operate only from gravity flow or storage tank water head), however, limited failures are assumed to occur for the purpose of considering the effects of flooding, spray, and wetting of equipment in the station analysis.

The identification of piping failure locations will be performed in accordance with Section 3.6.2.

3.6.1.1.2.1 Interaction Criteria

The following criteria define how interactions shall be evaluated.

1. Environmental Interaction

   An active component (electrical, mechanical, and instrumentation and control) is assumed incapable of performing its function upon experiencing environmental conditions exceeding any of its environmental ratings.

2. Jet Impingement Interactions

   Active components (electrical, mechanical, and instrumentation and control) subjected to a jet are assumed failed unless the active component is enclosed in a qualified enclosure, the component is known to be insensitive to such an environment, or unless shown by analysis that the active function will not be impaired.

3. Pipe Whip Interaction

   A whipping pipe is not to be considered to inflict unacceptable damage to other pipes of equal or greater size and wall thickness.

   A whipping pipe is only considered capable of developing through-wall leakage cracks in other pipes of equal or greater size with smaller wall thickness.

   An active component (electrical, mechanical, and instrumentation and control) is assumed incapable of performing its active function following impact by any whipping pipe unless an analysis or test is conducted to show otherwise.

3.6.1.1.3 Protective Measures

3.6.1.1.3.1 Reactor Coolant System

The fluid discharged from postulated pipe breaks at branch connections 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 or feedwater line break, propagation of damage from the original event is limited, and the components as needed, are protected and available.

For piping connected to the Reactor Coolant System (six inch nominal or larger) and all connecting piping out to the LOCA boundary valve (Figure 3-6) 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 is limited by piping separation and restraints so as not to 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. (See also paragraph 11.c of Section 3.6.2.1.2).
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.

Additional pipe restraint design criteria are discussed in Reference 1.

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 reactor vessel cavity wall, the refueling canal walls, various structural frames, rupture restraints, and physical distance limit the propagation of a break in any one reactor coolant loop from adversely affecting another loop from adversely affecting another loop or the

containment. The portion of the main steam and feedwater lines within the Containment has been routed behind the steam generators and their enclosures for facilitate separation from reactor coolant piping. The barriers described above are designed to withstand loadings resulting from jet and pipe whip impact forces.

Other than Emergency Core Cooling System lines, all 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 and then penetrate the crane wall in the vicinity of the loop to which they are attached.

3.6.1.1.3.2 All Other Mechanical Piping Systems

Preferred measures to protect against pipe whip, jet impingement and resulting reactive forces to assure plant safety 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, alternately, 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 rupture restraints and jet deflectors.
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.

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 condenser circulating water expansion joint, or the most severe Condensate System failure. 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.

The 6.9/4.16 KV transformers in the Turbine Building Basement have flood walls around them to protect against an RC pipe break. See Section 8.3.1.1.1.3.1 for more details.

In response to NRC IE Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," which cited a catastrophic failure of main feedwater pipe due to pipe wall thinning at Surry Power Station Unit 2, a description of the DPC erosion/corrosion monitoring program was submitted to the NRC for evaluation (letter from H.B. Tucker to NRC, dated September 14, 1987). Further DPC response on this issue was made resulting from the NRC issuing Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning,"on May 2, 1989. As committed to and described in the Duke Power Company response to Generic Letter 89-08 (letter from H.B. Tucker to NRC, dated July 21, 1989), Catawba Nuclear Station implemented the Pipe Erosion Control Program Manual within CNS Maintenance Management Procedure 3.0. The Pipe Erosion Control Program Manual has been superseded by PD-EG-ALL-1610, "Flow Accelerated Corrosion Program", and AD-EG-ALL-1610, "Flow Accelerated Corrosion Implementation". PD-EG-ALL-1610 gives an overall description of the Flow Accelerated Corrosion (FAC) Program and provides a roadmap to the scope, basis, and documents governing the program. AD-EG-ALL-1610 provides a standardized method of identifying, inspecting, and evaluating piping systems which are susceptible to FAC.

3.6.1.1.3.3 Main Steam And Feedwater System Design

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

1. Main Steam Lines are 100 percent cold pulled in the horizontal direction on Unit 1. Unit 2 main steam lines are not cold pulled.
2. Overpressure capability of the piping based on actual wall thicknesses is as follows:

	Normal Operating Pressure[6]	**System Design Pressure[7]**	**Margin**
Main Steam:	1006 psig	1185 psig	18%
Feedwater:	1144 psig	1385 psig	21%

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 Section III, Class 2.
4. For the Main Steam piping inside the S/G enclosure, a guard pipe is used in conjunction with a system of rupture restraints to limit the break opening area upon a postulated pipe break at the SG nozzle. This restricted break area is used to reduce the differential pressure associated with the S/G enclosure as described in Section 6.2.1.2 of the UFSAR. Jet impingement interactions also consider the restricted motion of the piping limited by the guard pipe and rupture restraints.
5. As a result of a Duke - NRC meeting in May 1976, guard pipe was removed from the main steam and main feedwater piping in the doghouse. These lines were originally designed with guard pipe. The main steam piping within the doghouse was identified as a break exclusion region. An augmented inservice inspection is performed for the piping within this break exclusion region in lieu of the guard pipe design as discussed in Section 6.6, for welds where no break is selected. Breaks in the main feedwater will be selected as outlined in Section 3.6.2.
6. Normal Operating Pressures for Main Steam and Feedwater reflect Unit 1 Measurement Uncertainty Recapture (MUR) power uprate (see Table 5-1 and Reference 20).
7. For system design pressures, refer to Figure 10-5 for Main Steam, and Figure 10-27 for Feedwater.

3.6.1.1.3.4 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 north and south sides by electrical penetration rooms which contain no piping. The east 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 west 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.

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

3.6.1.1.4 Acceptability Criteria

The capability to eventually achieve a cold shutdown condition is not jeopardized even if the pipe failure is followed by a single active failure. The system requirements and available redundancy are determined on a shutdown logic diagram, or a required equipment list for mitigating the effects of the postulated failure.

Repair of failures is considered to assure achievement of the cold shutdown condition where such repairs can be shown to be practical and timely, and provided the unit can be held in a safe shutdown state during the time required for the repair.

3.6.1.2 Description of Piping System Arrangement

Separation is the primary consideration in piping system layout and arrangement. Where physical separation is not feasible, protective devices are provided to protect essential components.

Table 3-17 provides a listing of high-energy systems. Moderate-energy systems are listed in Table 3-18. Control room habitability is discussed in Section 3.6.1.1.3.4.

An analysis of the effects on safety-related systems of failures in all high-energy or moderate-energy piping systems in accordance with the applicable criteria of the Standard Review Plan (SRP) Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment” was provided to the NRC by letter dated March 9, 1982 from W. O. Parker, Jr. to H. R. Denton. This analysis considered single active failures in accordance with BTP ASB 3-1. The Auxiliary Steam System, which is seismically supported inside the Auxiliary Building, was treated as moderate energy piping in the analysis.

Environmental qualification of equipment is discussed in Section 3.11.

3.6.1.3 Safety Evaluation

Safety functions are identified for each initiating event by the failure mode and effects analysis discussed in Section 3.6.2.1.2. For each postulated failure, every credible unacceptable interaction shall be evaluated. In establishing system requirements for each postulated break, it is assumed that a single active component failure occurs concurrently with the postulated rupture.

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

3.6.2.1 Criteria Used To Define Break And Crack Location And Configuration

3.6.2.1.1 Postulated Piping Break Location Criteria for the Reactor Coolant System

The design basis for postulated pipe breaks includes 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 in piping down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3-6) on outgoing lines[1] and down to and including the second check valve (Case III in Figure 3-6) on

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

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 assuming either of the two check valves in the line close.

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. Periodic testing is performed of the capability of the valves to perform their intended function. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3-6), 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, 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 10CFR 100 guidelines. These safety systems have been designed to provide protection for a Reactor Coolant System pipe 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 based on Emergency Core Cooling System analyses using realistic assumptions that 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 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 leak tightness is not decreased below the design value, if the break leads to a loss of reactor coolant[2].
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 does not cause a loss of reactor coolant or steam or feedwater line break, and
   b. A Reactor Coolant System pipe break does not cause a steam-feedwater system pipe break and vice versa.

In the unlikely event that one of the small (as defined above) pressurized lines should fail and additional in a loss of reactor coolant accident, the piping is restrained or arranged to meet the following criteria:

1. Break propagation must be limited to the affected leg (i.e., propagation to the other legs of the affected loop and to other loops is prevented).
2. Propagation of the break in the affected leg is permitted but is limited by piping separation and restraints to a total break area of 12.5 square inches (4 inch inside diameter). The

[2] The containment is defined here as the Containment vessel and pentrations, the steam generator shell, the steam generator steam side instrumentation connections, and the steam, feedwater, blowdown, and steam generator drain pipes within the Containment Structure.

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 is 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.

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 is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. The industry basis 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 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 Catawba 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 April 3, 1989). 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.

3.6.2.1.1.1 Postulated Piping Break Locations and Orientations

Reference 1 defines the original basis for postulating pipe breaks in the reactor coolant system primary loop. Reference 2 provides the basis for eliminating from certain aspects of design consideration previously postulated reactor coolant system pipe breaks, with the exception of those breaks at branch connections. See Table 3-20 and Figure 3-7. References 6, 18, and NRC Generic Letter 87-11 (Reference 19) provide the basis for eliminating arbitrary intermediate pipe rupture locations.

3.6.2.1.1.2 Postulated Piping Break Sizes

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

3.6.2.1.1.3 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 4 inches up to the largest connecting line. Where postulated, 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).

3.6.2.1.2 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 methods utilized to assure that concurrent single active component failure and pipe break effects do not jeopardize the safe shutdown of the reactor are outlined in Section 3.6.2.3.

2. Station general arrangement and layout design of high-energy systems utilize the possible combination of physical separation, pipe bends, pipe whip restraints and guard 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 eventually maintained in a cold 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, and vent openings.

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 used in the evaluation of the loss-of-coolant accident, steam line break accident, or feedwater line break.

    b. Offsite Power is assumed to be unavailable if a trip of the Turbine-Generator System or Reactor Protection System is a direct consequence of the postulated piping failure, (e.g., a loss-of-coolant accident, steam line break or feedwater line break).

    c. Seismic loadings equivalent to the Operating Basis Earthquake (OBE) are used in the analysis of piping systems for the purpose of postulating break locations. Protective structures are designed to withstand the effects of the postulated piping failure in combination with loadings associated with the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) within the respective design load limits for the structures.

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 that system only are not assumed, provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice 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 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 are 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 equal or 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. If such exception is taken, the analytical technique or experimental data used will be documented in the FSAR. The analytical technique for selected jet load evaluation is: a) jet loads on lines smaller (or larger with thinner walls) than the source line are calculated in accordance with Section 3.6.2.2.2.3, and, b) evaluated by including the jet loads in the target piping stress analysis.
11. Piping Breaks Within The LOCA Boundary (See Figure 3-6)
    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 that ECCS flow to other loops is maintained.
    c. For breaks in 6 inch nominal or larger piping, propagation of the break in the affected loop is not permitted if the resultant break area is more than 120 percent 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 pressures, loop hydraulic forces, reactor internals, reaction loads, primary equipment support loads, or ECCS performance. Propagation of the break is limited by the use of piping separation and restraints. (See also Section 3.6.1.1.3.1). Propagation to any other loop is not permitted in any case.

       A rupture of the piping between the pressurizer and the code safety and power operated relief valves results in release of saturated steam and is less severe than for an equal size break through which reactor coolant fluid discharges because of the higher specific volume of the steam. Because of this difference the following criteria apply to this specific case:
       1) Propagation of the break should not result in release of reactor coolant (liquid phase).
       2) Except for the upper level taps as discussed below, unlimited break propagation of the lines connected to the pressurizer above the normal water level is acceptable.
       3) Propagation of the rupture must not result in breakage of more than one upper level tap.
    d. Pipe breaks within the LOCA boundary that are equal to or less than 4 inch nominal pipe size must meet the criteria as outlined in Section 3.6.2.1.1.
12. Piping Breaks Outside the LOCA Boundary (Non-LOCA)
    a. A pipe break which is not a loss-of-coolant accident cannot cause a loss-of-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 as described in item 5 of this section.

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).

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.

e. A pipe break in a non-seismic system (Duke System Piping Class E, G, H) cannot result in damage to an essential system.

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 sub-compartment 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 safety shutdown and eventually maintain the reactor in a cold 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 eventually maintain the reactor in a cold shutdown condition.

16. Minimum essential component and systems performance is provided as required for the type of break.

17. The effects of pipe ruptures are not allowed to result in offsite doses in excess of 10CFR 100 allowable limits.

18. Operability in an environment is assured for all equipment required to mitigate the break by the equipment specification requirements based on conservative design conditions.

19. Emergency procedures are prepared that are to be followed after a postulated piping break for high-energy systems as required.

3.6.2.1.2.1 Postulated Piping Break Locations For High-Energy Piping Systems

Systems identified as containing high-energy piping are examined by detailed design drawing review for postulated pipe breaks as defined below. Systems analyzed for consequences of postulated piping breaks are listed in Table 3-17.

Terminal ends are considered as 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 intersections of runs of comparable size and fixity which have a significant

effect on the main run need not be considered terminal ends when the stress analysis model includes both the run and branch piping and the intersection is not rigidly constrained to the building structure.

The requirement to postulate arbitrary intermediate pipe breaks was eliminated by References 6, 18, and NRC Generic Letter 87-11 (Reference 19).

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 weld location of potential high stress or 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 shall not exceed 2.4 S_m except as noted below.
            b) The maximum stress range between any two load sets (including the zero load set) shall 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 S_m) but is not greater than 3 S_m, the limit of $U < 0.1$ shall be met.

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

               where:

S_n	=	primary-plus secondary stress range, as calculated from Equation (10) 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.

2. Breaks in Duke Class B and C 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 weld location of 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 and an OBE event, 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 I 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 ASME Boiler and Pressure Vessel Code, Section III.

3. To assure protection of safety-related structures, systems or components, breaks in Duke Class E, F, G and H 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 one of the following methods:
      1) For Class E, F, G, and H Piping:

         At each intermediate weld location of potential high stress or fatigue.
      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 the combination of loadings associated with the normal and upset plant condition loadings and an OBE event, 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, per ANSI B31.1.0.
S_A	=	allowable stress range for expansion stresses, per ANSI B31.1.0.

3.6.2.1.2.2 Postulated Piping Break Locations For Moderate-Energy Piping Systems

Systems identified as containing moderate-energy piping are examined by detailed drawing review for postulated through-wall cracks as defined below. Systems analyzed for consequences of postulated piping cracks are listed in Table 3-18.

1. Cracks in Duke Class B, C and F piping are postulated at the following locations:
   a. The terminal ends of the pressurized portions of the run.
   b. At intermediate pipe-to-fitting weld locations of potential high stress or fatigue (e.g. pipe fittings, valves, flanges and welded attachments) that result in the maximum effects from fluid spraying, flooding or environmental conditions except in portions of piping where the maximum stress range is less than 0.4 (1.2 S_h + S_A) as defined in items 2b2 and 3b2 of Section 3.6.2.1.2.1.
2. Cracks in Duke Class E, G and H piping are postulated at the following locations:
   a. The terminal ends of the pressurized portions of the run.

b. At intermediate pipe-to-fitting weld locations of potential high stress or fatigue (e.g. pipe fittings, valves, flanges and welded attachments) that result in the maximum effects from fluid spraying, flooding or environmental conditions.

3.6.2.1.2.3 Postulated Break Type, Size, and Orientation

1. Circumferential Pipe Breaks

   The following circumferential breaks are postulated in high-energy fluid system piping at the locations specified in Section 3.6.2.1.2.1.

   a. Circumferential breaks are postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range exceeds the limits of Section 3.6.2.1.2.1, items 2b2 and 3b2 but the circumferential stress range is at least 1.5 times the axial stress range.

   b. Where break locations are selected in fittings in accordance with Section 3.6.2.1.2.1 without the benefit of detailed stress calculations, breaks are postulated at each weld, in piping greater than one inch NPS, to the fitting, valve, or welded attachment. Alternately, a single break location at the section of maximum stress range may be selected as determined by detailed stress analyses or tests on a pipe fitting.

   c. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints. No limited break areas will be used for compartment pressurization calculations. If limited break areas are used for jet impingement reviews, the basis will be the installation of rigid rupture restraints. Exceptions to this criteria are related to Main Steam piping inside containment as described in Section 3.6.1.1.3.3.

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

   e. Postulated pipe whip for target review will be defined by engineering judgement based on piping geometry, jet thrust direction, break location analysis type, and hanger location and type. When further confirmation is required, postulated piping breaks and targets are field reviewed after the drawing based analysis has been completed. For the purposes of analysis, breaks are assumed to reach full opening size in one millisecond after break initiation.

2. Longitudinal Pipe Breaks

   The following longitudinal breaks are postulated in high-energy fluid system piping at the locations specified in Section 3.6.2.1.2.1.

   a. Longitudinal breaks in fluid system piping and branch runs are postulated in nominal pipe sizes 4 inches and larger, except where the maximum stress range exceeds the limits of Section 3.6.2.1.2.1, items 2b2 and 3b2 but the axial stress range is at least 1.5 times the circumferential stress range.

   b. Longitudinal breaks are not postulated at terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds.

c. Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented perpendicular above and below the plane of whip (but not concurrently) at two diametrically-opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternately, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

d. The dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

e. Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis. For the purpose of analysis, breaks are assumed to reach full size one millisecond after break initiation.

3. Through-Wall Leakage Cracks

The following through-wall leakage cracks should be postulated in moderate-energy fluid system piping at the locations specified in Section 3.6.2.1.2.2.

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

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

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

d. Cracks are not postulated in portions of Duke Class B, C, or F piping where the stresses are less than 0.4 (1.2 S_h + S_A). Through-wall cracks are not postulated inside containment because environmental consequences are enveloped by high energy circumferential breaks.

3.6.2.1.3 Failure Consequences Associated With Postulated Pipe Breaks

The interactions that are evaluated to determine the failure consequences are dependent on the energy level of the contained fluid. They are as follows:

1. High-Energy Piping

a. Circumferential Breaks and Longitudinal Splits

1) Pipe Whip (displacement)

2) Jet Impingement

3) Compartment Pressurization

4) Flooding

5) Environmental Effects (Temperature, humidity, water spray)

b. Through-wall leakage cracks

1) Environmental Effects (Temperature, Humidity, water spray)
2) Flooding

2. Moderate – Energy Piping
   a. Through-wall leakage cracks
      1) Flooding
      2) Environmental Effects (Temperature, humidity, water spray)

For high energy piping there are certain exceptions as detailed in Reference 1 for the reactor coolant loop.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Reactor Coolant System Dynamic Analysis

3.6.2.2.1.1 Westinghouse Scope

This section summarizes the dynamic analysis as it applies to the LOCA resulting from the postulated design basis pipe breaks at main reactor coolant branch line interconnections. 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 Reference 1 as it pertains to postulated breaks at branch connections.

The particular arrangement of the Reactor Coolant System for the Catawba Nuclear Station is accurately modeled by the standard layout used in Reference 1 and the postulated branch connection break locations do not change from those presented in Reference 1.

In addition, an analysis is 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 are based on full pipe area discharge except where limited by major structures.

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 is 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 3-9.

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 deflections at mass points 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 connection. The general dynamic solution process is shown in Figure 3-10.

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 (LOCA) as a result of 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-IV (Reference 4) code was developed with a capability to provide this information.

The SATAN-IV 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; (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 including the W-3 DNB correlation in addition to the reactor kinetics are incorporated in the code.

The SATAN-STHRUST computer program was developed to compute 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:

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

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

ρ = Density, Lb_m/Ft^3

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

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 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 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 Reference 1.

3.6.2.2.1.2 Reactor Coolant System Dynamic Analysis for Steam Generator Replacement (Unit 1)

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 3-301 through 3-304).

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 8 in Section 3.6.3) 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:

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

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

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 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.

3.6.2.2.2 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, compartment pressurization, and temperature effects establish the need for pipe whip restraints.

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 do not prevent mitigation of the break nor prevent the safe and continued shutdown of the reactor.

3.6.2.2.2.1 Assumptions

1. The thrust load acting on the pipe due to a blowdown jet is equal and opposite to the jet load.
2. The discharge coefficient is equal to 1.0.
3. The break opens to its defined size in 1 millisecond.
4. For the purpose of estimating jet forces, the blowdown shall be to an infinite volume at standard ambient conditions.
5. The initial fluid condition within the pipe prior to rupture is that for the worst case normal plant operating condition.
6. The jet profile expansion half-angle is 10 degrees.

3.6.2.2.2.2 Blowdown Thrust Loads

The thrust force at any time, T (t) is given by

$$T(t) = \left(\frac{\rho E V_E^2}{g_c} + \{P_E - P_A\} \right) A_{jE}$$

where:

ρE = fluid density at break at time t

V_E = fluid velocity at break at time t

A_{jE} = pipe break exit area

P_E = control volume pressure at break at time t

P_A = ambient pressure

g_C = gravitation constant

A simplified analysis may be conducted by assuming that the fluid is blowing down in a steady-state condition with frictionless flow from a reservoir at fixed absolute pressure P_o (P_o is the initial line pressure). When the fluid is subcooled, nonflashing liquid, the flow will not be critical at the break area so that $P_E = P_A$ and $V_E = \sqrt{2g_c \ (P_o - P_A) / \rho_E}$. If $P_A << P_o$ the thrust force may be conservatively approximated by

$$T = 2P_o A_{jE}$$

When the fluid is saturated, flashing or super-heated vapor, the fluid can be assumed to be a perfect gas. The velocity for critical flow at the break area is given by

$$V_E = (Kg_c P_E / \rho_E)^{1/2}$$

and

$$P_E = P_o (\frac{2}{k+1})^{K/K-1}$$

where

K = C_p/C_v is a ratio of specific heats

C_p = specific heat at constant pressure

C_v = specific heat at constant volume

A value of K=1.26 is justified for steam as being conservative. If $P_E >> P_A$, the thrust force may be conservatively approximated by:

$$T = 1.26 \ P_o A_{jE}$$

3.6.2.2.2.3 Jet Impingement Loads

The loads on an object exposed to the jet from a pipe break can be determined from the blowdown thrust and the profile of the impinged object.

$$Y_j = T \bullet \frac{A_i}{A_j} \bullet S_F \bullet D_{LF} \cos\phi$$

where

Y_j	=	Normal load applied to a target by the jet
A_i	=	Cross-sectional area of jet intercepted by target structure
A_j	=	Total cross-sectional area of jet at the target structure
S_F	=	Shape factor
D_{LF}	=	Dynamic load factor
T	=	Total blowdown thrust at break as calculated in Section 3.6.2.2.2.2.
ϕ	=	Angle between jet axis and the target.

The ratio A_i/A_j represents the portion of the total mass flow from the jet which is intercepted by target structure. A dynamic load factor of 2.0 shall be used in the absence of an analysis justifying a lower value.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 General Criteria for Pipe Whip Evaluation

1. The dynamic nature of the piping thrust load shall be considered. In the absence of analytical justification, a dynamic load factor of 2.0 is applied in determining piping system response.
2. Elastic-perfectly plastic pipe and crushable material properties may be considered as applicable. Consideration for crushable materials is described in Table 3-22.
3. Pipe whip is considered to result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the thrust force. A maximum of 180° rotation may take place about any hinge.
4. The effect of rapid strain rate of material properties is considered. A 10 percent increase in yield strength is used to account for strain rate effects.

3.6.2.3.2 Analysis Methods

The pressure time history, jet impingement load on targets, and the thrust resulting from the blowdown of postulated ruptures in piping systems is determined by thermal and hydraulic analyses or conservative simplified analyses.

In general, the loading that may result from a break in piping is determined using either a dynamic blowdown or a conservative static blowdown analysis. The method for analyzing the interaction effects of a whipping pipe with a restraint will be one of the following:

1. Equivalent static method
2. Lumped parameter method
3. Energy balance method

In cases where time history or energy balance method is not used, a conservative static analyses model will be assumed.

The lumped parameter method is carried out by utilizing a lumped mass model. Lumped mass points are interconnected by springs to take into account inertia and stiffness properties of the system. A dynamic forcing function or equivalent static loads may be applied at each postulated break location with unacceptable pipe whip interactions. Clearances and inelastic effects are considered in the analyses.

The energy balance method is based on the principle of conservation of energy. The kinetic energy of the pipe generated during the first quarter cycle of movement is assumed to be converted into equivalent strain energy, which is distributed to the pipe or the support. The strain in the restraint is limited to 50 percent of the ultimate uniform strain.

3.6.2.3.3 Pipe Whip Restraint Design

When required, restraints are designed to protect essential components from the dynamic effects of pipe whip and jet impingement. The loadings on the restraint are determined by one of the methods outlined in Section 3.6.2.2 and Section 3.6.2.3. The design of these restraints follows the guidelines of AISC (Ref. 5); however, since pipe rupture is associated with the faulted plant condition, higher stress allowables are permitted as identified in Table 3-22. Where a restraint is also designed to function as a piping support, the discussion in Section 3.9.3.1.5 is applicable. Rupture loads with a dynamic load factor of 2.0 shall be added to the faulted loads and the support designed for faulted condition per Table 3-101.

3.6.2.4 Mechanical Penetrations

Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of ASME Section III, Subsections NC and NE 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 Class C through H lines are upgraded to Class B between Containment isolation valves).

The process line making up the pressure boundary is consistent with the system piping materials, fabrication, inspection, and analysis requirements of ASME Section III, Subsection NC.

Critical high temperature lines and selected engineered safety system and auxiliary lines (regardless of temperature) require the "Hot Penetration" assembly as shown on Figure 3-13 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 (0.5 percent damping) as developed from the method described in Section 3.7.2 and enveloped for conservatism. Loading combinations and stress criteria for penetrations are shown in Table 3-21. The design of guard pipes considers the simultaneous effects of pressure and jet loadings resulting from a rupture within the guard pipe and the SSE loadings.

3.6.2.4.1 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 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 primary and secondary penetrations are designed to maintain Containment integrity for any loss-of-coolant accident combination of Containment pressures and temperatures.
2. Quality assurance measures for penetration design calculations, criteria, documentation and procedures are in accordance with the design control requirements of Chapter 17.
3. 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 as appropriate.
4. 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, Subsections NC and NE.
    b. All mechanical penetrations and their anchorages are analyzed in accordance with the requirements of ASME Section III, Class 2, Subsection 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.

3.6.2.4.2 Hot Penetrations

Typical hot penetration assemblies as shown on Figure 3-13 consist 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 follow:

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 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 so as to act as a single unit under thermal, seismic, and pipe rupture loads.
6. Stress levels for process lines meet requirements of Section 3.9.3.
7. Stress levels for guard pipes and other penetration structural components meet the requirements of Section 3.9.3.
8. Exterior bellows cover and impingement plate protects the bellows assembly from foreign objects during construction and station operation.

9. The process pipe is designed to meet the requirement of Table 3-96 for stress levels and applicable loading combinations. The process pipe is of seamless construction made from SA376 TP304 or TP316 stainless steel, except for Main Steam and Main Feedwater penetrations which are SA-106 Gr B or Gr C. (See Figure 3-12)

Design codes applicable to hot penetrations are as described below.

1. Penetration boundaries are in accordance with ASME III, Subsection NE, Paragraph NE-1100. Process lines including flued head, guard pipe, and bellows assemblies including dished heads, are designed, fabricated, and inspected to ASME III, Subsection NC, with the allowable stresses as defined above. In addition the guard pipe wall thickness design complies with the requirements of NE-3324.3a of the ASME Code when using the design pressure and temperature of the enclosed process pipe.
2. The Reactor Building anchor section is considered a structural component. Attachment welds to the guard pipe meet and are inspected to ASME Section III, Subsection 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 Section III, Subsection NE.

3.6.2.4.3 Residual Heat Removal Recirculation Line Penetration

Residual heat removal recirculation line penetrations are of the cold-penetration type. (See Figure 3-11)

Design requirements for these penetrations are as follows:

1. The recirculation line is an extension of Containment up through the first valve.
2. These valves are Safety Class 2 and are conservatively designed (600 psig design pressure) to withstand the Containment design pressure of 15 psig.
3. Valves are located in an accessible area for maintenance during the post-accident period.
4. Expansion joints are utilized in the penetration design.

The stress analysis for the two recirculation lines between containment and their sump isolation valves shows that the stress in these sections is below values that would require postulation of breaks occurring if this pipe were normally in use (i.e., stress does not exceed 0.4 ($1.2S_H + S_A$). For this reason it is acceptable not to provide guard pipe for these sections of piping (see SRP 6.2.4 Acceptance Criteria 6.e.). Any postulated leakage from the valve bonnet via the valve stem is contained via a leakoff that is directed to the Recycle Holdup Tank (the RHT is equipped with a diaphragm that would contain any gases released from solution). The valves themselves are design to withstand 600 psig at 400°F which is well in excess of values to which they would be exposed after an accident (approximately 36 psig, 190°F).

3.6.2.4.4 Access for Periodic Examination

A description of the method of providing access to permit periodic examinations of process pipe welds within the protective assembly as required by the plant inservice inspection program is discussed in Section 6.6.

3.6.2.5 Summary of Dynamic Analyses Results

A summary of postulated Unit 1 circumferential break locations was shown for illustrative purposes on former FSAR Figures 3.6.2-9 through 3.6.2-207 (Figure 3-14 through Figure 3-212

in reformatted FSAR). Only Unit 1 results were presented, however with the exception of the differences allowed by References 6 and 7, the methods and result associated with Catawba 2 are very similar to those of Catawba 1. Since the former figures were for illustrative purpose only, they are no longer included and are available for review in Duke Power licensing files. Complete controlling documentation is maintained in Duke Power Engineering calculation files.

In addition to the pipe routing, postulated break locations and pipe whip restraints shown on the aforementioned figures, a summary of Unit 1 break location and protective requirements was presented in former FSAR Table 3.6.2-4 (Table 3-23 in reformatted FSAR). Again, only Unit 1 results were presented, however with the exception of the differences allowed by References 6 and 7, the methods and results associated with Unit 2 are very similar to those of Catawba 1. Since the former table was for illustrative purposes only, it is no longer included and is available for review in Duke Power licensing files. Complete analysis and results are maintained in Duke Power Design Engineering calculation files, while configuration control documentation is maintained in controlled engineering drawings.

3.6.3 References

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2. Letter from H.B. Tucker (DPC) to E.G. Adensam (NRC), dated May 11, 1984, transmitting Westinghouse report justifying elimination of RCS loop pipe breaks from certain primary design considerations.
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7. "Request for Exemption from a Portion of GDC 4 of Appendix A to 10CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as a Structural Design Basis for Catawba Nuclear Station, Unit 2", transmitted by letter dated April 23, 1985 from E. G. Adensam (NRC) to Hal B. Tucker (Duke).
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9. 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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12. Duke Power Company, Letter from H.B. Tucker to NRC, April 3, 1989, re: Response to NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems."

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15. Nuclear Regulatory Commission, Letter to All Licensees for Nuclear Power Plants Holding an Operating License or a Construction Permit, from James G. Partlow, May 2, 1989, "Erosion/ Corrosion-Induced Pipe Wall Thinning (Generic Letter 89-08)."

16. Duke Power Company, Letter from H.B. Tucker to the NRC, September 14, 1987, "Catawba Nuclear Station, Docket Nos. 50-413, 414; McGuire Nuclear Station, Docket Nos. 50-369, 370; Oconee Nuclear Station, Docket Nos. 50-269, 270, 287; IE Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants."

17. Duke Power Company, Letter from H.B. Tucker to the NRC, July 21, 1989, "Oconee Nuclear Station, Docket Nos. 50-269, 270, 287; McGuire Nuclear Station, Docket Nos. 50-369, 370; Catawba Nuclear Station, Docket Nos. 50-413, 414; Response to Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning".

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3.7 Seismic Design

Note:

This section of the FSAR contains information on the design bases and design criteria for seismic design and analysis. Additional information that may assist the reader is contained in the Plant Design Basis Specification for Seismic Design (CNS-1465.00-00-0007).

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 Section 2.5.2. The spectra defined in Section 2.5.2. are the response at top of sound rock.

Figure 2-110 (Part 3) gives the smoothed spectra for the OBE at two percent damping. The amplification of base motion to the peak response is approximately 3.5 from 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.

The design ground response spectra obtained from the procedures outlined in Regulatory Guide 1.60 are plotted against the spectra used for the design of Catawba in Figure 3-213.

The curves of interest are the 2% and 5% damping curves. The 2% curve is used for welded steel structures and the 5% curve is used for concrete structures. The lowest first mode frequency of any Category I structure is 5.4 Hz. Therefore, comparison is made from the 5.4 Hz frequency level and higher.

For the 5% damping curve, the Catawba design acceleration spectra envelopes or equals that resulting from the Regulatory Guide 1.60 criteria.

In the frequency range above 12.5 Hz on the 2% damping curve, the Catawba design acceleration spectra equals that resulting from the criteria of Regulatory Guide 1.60. Between 12.5 Hz and 7.7 Hz on the 2% curve, the Regulatory Guide criteria envelope the Catawba design criteria. (The maximum discrepancy in this frequency range is 5% with the average difference less than 2.5%). In the frequency range between 7.7 Hz and 5.4 Hz, the Catawba design acceleration spectra again envelopes that resulting from the Regulatory Guide 1.60 criteria.

Compliance with Regulatory Guide 1.60, Design Spectra for Seismic Design of Nuclear Power Plants is discussed in Section 1.7.

3.7.1.2 Design Time History

All Category I 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. The numerical average of the response of the four synthetic earthquakes at the respective period coordinate is used to establish the final design

spectra at a specified elevation in the support structure. Figure 2-110 provides 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 are 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
	125	25	2
	25	4	0.3
	4	1	0.15
CASE III.	FROM	TO	STEP SIZE
	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 is used to establish the response spectra used for comparison.

As can be seen from the comparison of Cases I and II in Figure 3-310, the step sizes in Case II produced additional peaks in the 0.1 to 0.25 second range.

As shown in the comparison of Cases II and III in Figure 3-214, the step sizes used in Case II 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 3-215.

An evaluation of Figure 3-310, Figure 3-214, and Figure 3-215 indicates that the results of Case III represent all peaks calculated by Case I and Case II. An evaluation of the shape of Class 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, except when the direct generation method is used.

When the direct generation method is used, a minimum of 16 frequency points per octave (i.e., the range of frequencies, $f_1 \leq f \leq f_2$, where $f_2 = 2f_1$) in addition to significant building frequencies are calculated.

3.7.1.3 Critical Damping Values

The following damping values are used for the seismic design of Category I structures, systems, and components for the OBE and SSE, unless higher damping factors can be justified.

ITEM	PERCENT CRITICAL DAMPING	OBE	SSE
Small-diameter piping systems, diameter equal to or less than 12 in.[1,2]		1.0	2.0
Primary Coolant Loop System piping (Unit 1)[3]		N411	N411
Primary Coolant Loop System equipment (Unit 1)[3]		2.0	3.0
Primary Coolant Loop System Components		0.5	1.0
Large-diameter piping system (diameter greater than 12 in.)[1,2]		2.0	3.0
Equipment and Components		2.0	2.0
Containment Vessel		1.0	2.0
Welded Steel Structures		2.0	2.0
Bolted Steel Structures		5.0	5.0
Control Rod Drive Mechanisms		5.0	5.0
Concrete Structures		5.0	5.0
Fuel Assemblies		7.0	10.0

Notes:

1. As an option to the damping values listed above for piping, an alter native set of values may be used as follows:

   5% damping for piping frequencies below 10 hz.; the % damping linearly decreasing to 2% at 20 hz. and 2% damping for frequencies above 20 hz. These values for damping apply to all pipe sizes for both Safe Shutdown Earthquake (SSE) and Operational Basis Earthquake (OBE) loadings. No combination of the two damping criteria are used for an analysis and the alternative set of damping values are not used in a time history analyses. Figure 3-216 illustrates the alternative set of damping values.

2. For the independent support motion methodology, (See Section 3.7.3.1.1.1) the damping for piping is limited to that specified in the table above. Alternative damping valves as described in Note 1 are not applicable.

3. Unit 1 damping values for RCL piping are per ASME Code Case N-411. Damping values for Unit 1 RCL equipment are per Regulatory Guide 1.61.

The stress levels in structural elements are not the same for all the elements of a whole structure; therefore, a single damping value cannot be accurately assigned to a total structure based upon a single stress level. The damping values listed above are average values based upon lower than average stress levels in the structure.

The damping values as tabulated are less than the referenced values in Regulatory Guide 1.61 for the higher stress conditions which correspond to the SSE. The tabulated damping values are less than or equal to the referenced lower stress values (OBE) except for concrete and bolted steel structures. An assessment was made to determine the impact should Catawba be required

to conform to the lower damping values recommended by Regulatory Guide 1.61. This guide recommends damping values of 4% and 7% for OBE and SSE load cases, respectively, when designing concrete and bolted steel structures. Catawba designs utilize damping values of 5% for both earthquake load cases. Evaluations indicate that a stress increase of approximately 7% would be realized in the OBE load condition if the 4% damping values are used in lieu of the 5% values. The SSE load condition, however, controls the design of our concrete and steel structures. The SSE load cases create stresses approximately 85% higher than the OBE load cases. Since the controlling SSE design load case utilizes a more conservative damping value than that recommended by Regulatory Guide 1.61, conformance to this guide would have no impact on the Catawba design.

Tests on fuel assembly bundles justify conservative component damping values of 7.0 percent for OBE and 10.0 percent for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests is found in Reference 9

The damping values used in component analysis of CRDM's and their seismic supports were developed by testing programs performed by Westinghouse. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie rods to the refueling cavity wall. The test conducted was on a full size CRDM complete with rod position indicator coils, attached to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support.

The program consisted of transient vibration tests in which the CRDM was deflected a specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variation in the drive shaft axial position, upper seismic support clearance and initial deflection amplitude was investigated.

Variation of the upper support clearance had the largest effect on the CRDM damping with the damping increasing with increasing clearance. With the upper clearance of 0.06 inches, the measured damping was approximately 8.0 percent. The clearances in a typical upper seismic CRDM support are a minimum of 0.10 inches. The increasing damping with increasing clearances trend from the test results indicates that the damping would be greater than 8.0 percent for both the OBE and SSE based on a comparison between typical deflection during these seismic events to the initial deflections of the mechanisms in the test. Component damping values of 5.0 percent are, therefore, conservative for both OBE and SSE.

3.7.1.4 Supporting Media for Seismic Category I Structures

The supporting media for each Seismic Category I structure are defined in Section 2.5. For information regarding total structural height see Figure 1-8 and Figure 1-9.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Category I Structures

3.7.2.1.1.1 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 superimposing 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 inertial 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 (Reference 15) 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 (References 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 displacement corresponding to a natural frequency, at any prescribed point on the meridian of the shell, resulting from a given loading condition.

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:

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

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.

W_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 W_i for the i^{th} mode.

N = Number of modes considered.

Equipment located in a structure is used as input to the seismic analysis. The equipment is comparatively rigid by virtue of its mass when compared to the mass of the structure. The equipment is connected to the structure in a rigid fashion.

The equipment masses and locations are used as input to the center of mass calculation for an individual location where mass will be lumped for use in the seismic model. The individual mass points are connected to the seismic model by rigid members and response spectra are generated for each point. The response spectra as generated are then used in the individual equipment design.

3.7.2.1.1.2 Containment Interior Structure, Nuclear Service Water Intake Structure, and Auxiliary Building, Including Diesel Building, Exterior Doghouse, and Upper Head Injection Building

The seismic loads on the Containment Interior Structure, Nuclear Service Water Intake 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 procedure used to lump masses for the seismic structural model is dependent upon the actual mass distribution and structural characteristics of the structure. The lumped parameter model used in the seismic analysis of the Catawba Nuclear Station, Reactor Building interior structure contained only the masses of the NSSS components. The stiffness characteristics of the individual items were not used in the model. This method of treating these components was state of the art at that point in time and was accepted when the PSAR was submitted. In spite of the above, additional work was undertaken. (See the response to SEB Action Item 5, which was transmitted by letter of April 8, 1982, W. O. Parker, Jr. to H. R. Denton). The results of this work indicate that if the NSSS System were to be included in the structural model, two basic changes would result. One, the systems force response would drop from 10% to 15%, indicating that the building is adequately designed and two, the resulting response spectra would decrease but would show a frequency shift of from approximately 10% to 15% toward the low frequency end of the curve. The resulting spectra should fall within the present spectra. Therefore, it is concluded that the simplification used in the original calculation is justified.

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

Sufficient mass locations of uniform distribution are established for the Diesel Building and Nuclear Service Water Pump Structure to assure consideration of all significant modes with frequencies less than $20H_Z$. Structures with fundamental frequencies greater than $20H_Z$ are designed as rigid structures with a constant acceleration equal to the acceleration corresponding to $20H_Z$ on the response spectrum.

The mass of the equipment is lumped at the 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 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 3.7.2.5 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 Section 3.7.3.8.

The mass of vertical structural members is distributed to the adjacent mass locations. Between these mass locations, vertical structural member properties are calculated for moments of inertia, cross-sectional area, effective shear area, and length. For those elevations where the center of mass and center of structural rigidity do not coincide or where the centers of structural rigidity change locations, infinitely stiff horizontal members connect these points.

All translational and rotational degrees of freedom are considered for each mode except the base which is fixed.

With the geometry and properties of the model defined, the model's influence coefficients (the flexibility matrix) are determined. 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 by solving the following characteristic equation:

$$\underline{K} - W_n^{\ 2} \underline{M} \, \phi_n = \{\underline{0}\}$$

where,

$\underline{K}$ = Stiffness matrix

$\underline{M}$ = Diagonal mass matrix

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

$\underline{0}$ = Zero vector

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

Having obtained the frequencies and mode shapes and employing the appropriate damping factors, 3.7.1.3, the spectral acceleration for each mode can be obtained from Design Ground Motion response spectra curves as defined in Section 2.5.2. 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:

$$A_{ij} = \gamma_j \, \phi_{ij} \, S_{aj}$$

where,

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 jth mode as obtained from Section 2.5.2

The response displacement may be obtained by:

$$D_{ij} = \gamma_j \, \phi_{ij} \, S_{dj}$$

where,

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

= S_{aj} / w_j^2

W_j = Natural circular frequency for the j^{th} mode.

D_{ij} = Response displacement at mass point i, for mode j

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

$$q_{ij} = M_i A_{ij}$$

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

$$V_{ij} = \sum_{Y=1}^{N} q_{yj}$$

and the effective moment is

$$M_{ij} = \sum_{Y=j}^{N} q_{yj} x_y$$

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 y

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,

$$R_i = \left[\sum_{j=1}^{N} R^2{}_{ij} \right]^{1/2}$$

N = Number of modes considered

R_i = Structural response such as acceleration (A), displacement (D), force (q), shear (V), or moment (M) at mass point i.

Significant mode shapes are determined by limiting the number of modes to those with frequencies less than 20 Hz. Structures with fundamental frequencies greater than 20 Hz are designed as rigid structures with a constant acceleration equal to the acceleration corresponding to 20 Hz on the response spectrum. An alternate method used for the Containment Interior Structure determines the number of significant modes by investigating the individual contributions of each mode as illustrated in Table 3-25. It is seen that beyond the first mode there is no significant contribution to the total (SRSS) response from the higher modes. However, a minimum of four modes are taken for all interior structural analysis.

Figure 3-217 through Figure 3-221 show sketches of mathematical models used for Category I structures.

Standard Review Plan 3.7.2, Seismic Systems Analysis, Part III, Section II, states that an acceptable method for accounting for accidental torsion is to add an additional 5% of the maximum building base dimension to the eccentricity that exists naturally in the building. This was not done at Catawba because the requirement did not exist at the time of the initial analysis.

The effect of incorporating this requirement into the Catawba analysis has been evaluated using the Auxiliary Building as the case study. The Auxiliary Building was selected because it exhibits the largest existing eccentricities as well as the largest plan dimensions.

The resulting increase in moment arm length ranged from 72% to 738% of the natural eccentricity. This increase in moment arm length increased the torsional moment from 24% to 135% over the original analysis results. Additionally, the distribution of this torsion and shear would vary over what was originally designed. Not only are the total torsional values higher, but the higher elevations of the structure would experience higher proportions.

It is Duke's position that the Catawba seismic model accurately depicts exact locations of centers of mass and rotation; and, therefore the 5% conservative increase is not required.

3.7.2.1.1.3 Fuel Pool and Fuel Handling Building

The seismic loads on the Fuel Pool and Fuel Handling Building as a result of base excitation are determined by a dynamic analysis. The dynamic analysis is made by modeling the structure as a space frame with joints connected by elastic members and finite elements. The base joints of the structure are considered fixed.

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

1. The formulation of a mathematical model consisting of joints connected with elastic members and finite elements. The choice of the location of these members and elements depends on their actual geometrical configuration in the structure. For each member and element in the structure, properties are calculated for moments of inertia, cross-sectional area, effective shear area, and length.
2. The derivation of the model's influence coefficients (the flexibility matrix). The contributions of flexure, as well as shearing deformations are considered. The number of simultaneous equations to be solved to obtain eigenvalues and eigenvectors is reduced by kinematic condensation of dependent degrees of freedom. The number of degrees of freedom used is at least greater than 6 times the number of desired mode shapes. The selection of joints chosen as independent is based on an intuitive approach.

The resulting flexibility matrix is inverted to obtain the stiffness matrix, which is used together with the mass matrix to obtain the eigenvalues and associated eigenvectors as outline in Section 3.7.2.1.1.

The standard response spectrum technique and the combination of modal contributions by the square root of the sum of the squares (SRSS) are used to determine structural response as defined in Section 3.7.2.1.1.

Significant mode shapes are determined by limiting the number of modes to those with frequencies less than 20 Hz. Structures with fundamental frequencies greater than 20 Hz are designed as rigid structures with a constant acceleration equal to the acceleration corresponding to 20 Hz on the response spectrum.

The horizontal acceleration of the fuel pool generates horizontal hydrodynamic forces acting outward on one side of the fuel pool and inward on the opposite side. A resultant force, P, is created which tends to translate the fuel pool horizontally. This force, which is the sum of the dynamic fluid forces acting on each side of the pool, is numerically equal to the horizontal shear on a section just above the bottom of the fuel pool.

The force P acts on the pool at some distance above the bottom and creates a bending moment at a section just above the bottom. This moment is resisted by a vertical couple consisting of compressive fiber stresses in the fuel pool wall on the side that resists outward forces and tensile stresses in the wall on the opposite side. This procedure for modeling the hydrodynamic phenomena for the fuel pool building is further described in Reference 18.

3.7.2.1.2 Category I Systems and Components

For a description of the seismic analysis methods used for Category I systems and components, see Section 3.7.3.

3.7.2.2 Natural Frequencies and Response Loads

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

1. The Reactor Building

   a. Natural Frequencies and Mode Shapes

      Figure 3-222 illustrates the first and second horizontal mode shapes and associated natural frequencies of the Reactor Building.

      The first and second vertical mode shapes and associated natural frequencies of vibration of the Reactor Building are shown in Figure 3-223.

   b. Response Loads

      The response loads of the Reactor Building due to the safe shutdown earthquake (SSE) are shown in Figure 3-224 through Figure 3-229. 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 3.7.2.1.1.1 for more details on the seismic analysis of the Reactor Building, and the method of combining the individual modal responses.

2. The Containment Interior Structures

   The mathematical model of the Containment Interior Structure is shown in Figure 3-217. The seismic analysis procedure is fully outlined in Section 3.7.2.1.1. Some of the numerical results include:

   a. North-South

      The first four horizontal mode shapes and associated natural frequencies of vibration of the Containment Interior structure are shown in Figure 3-230.

   b. East-West

      For this direction, the first four horizontal mode shapes of the interior structure and the associated natural frequencies are shown in Figure 3-231.

   c. Vertical

      Figure 3-232 illustrates the first two vertical mode shapes of vibration and their associated natural frequencies of the Containment interior structure.

   d. 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 3.7.2.1.1. 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 3-233 for the North-South direction SSE and in Figure 3-234 for the East-West direction SSE.

Figure 3-235 through Figure 3-240 illustrate the response spectra of the interior structure at important equipment elevations as well as at other critical points of support.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Structural Modeling

Axisymmetric structures including the Reactor Building and Containment Vessel are modeled using Kalnins' computer program for axisymmetric shells of revolution. The mathematical model of this computer program is described in Section 3.7.2.1.1.1.

More irregular structures including the Containment interior structure and the Auxiliary Building are represented as three dimensional stick models with lumped masses at the nodes. A detailed description of these models is given in Section 3.7.2.1.1.2.

All models are considered fixed at the base where they rigidly connect to a massive reinforced concrete foundation on solid rock or fill concrete extending to solid rock. All structural elements with no interconnection other than through the foundation are modeled as a separate and complete structural system. No interaction is considered for separate structural systems supported on the same foundation. See Section 3.7.2.4 for a more detailed discussion.

3.7.2.3.2 Component and System Modeling

For a description of the criteria and procedures used for component and system modeling, see 3.7.3.

3.7.2.4 Soil/Structure Interaction

All major Category I structures are founded on solid rock and/or fill concrete extending to solid rock as discussed in Section 3.7.1.4. Since no major Category I structures are supported on soil there is no variation in structural behavior as a result of soil/structure interaction. Interaction of adjacent structures founded on sound rock at the Catawba Nuclear Station will not result in a significant variation from the "free field" seismic condition. This is verified for the Reactor Building of the McGuire Nuclear Station which is identical to Catawba Nuclear Station. (This is documented in McGuire FSAR Section 3.7.2.4).

The design of substructure walls of the Auxiliary Building and several associated structures with respect to seismic lateral earth pressure was done by a conservative interpretation of recommendations in Reference 17. A horizontal pressure resulting from 18% of the weight of a wedge of soil was used which produced an incremental seismic lateral earth pressure of 61% of the static at rest earth pressure. This incremental pressure was applied at a height of two-thirds of the distance above the base of the wall and added to the static (at rest) earth pressure.

For comparison with another state-of-the-art method, soil pressure was calculated using a Mononobe-Okabe seismic coefficient analysis taken from Reference 25. Using this reference K_{AE} which is a total (static and seismic) coefficient was calculated. Then after calculation of K_A, the corresponding active earth pressure coefficient, an incremental seismic lateral coefficient

$\frac{\Delta K_{AE}}{K_A}$ was then applied to the at rest earth pressure coefficient, K_o, and the resultant of the pressure was conservatively applied at a point two-thirds of the wall height above the base.

As is shown by the soil pressure diagrams for the Auxiliary Building in Figure 3-241, the original design for this structure is conservative compared to the results for the Mononobe-Okabe seismic coefficient analysis.

All other Category I structures have been checked using the Mononobe-Okabe analysis and were found to have been designed conservatively. Soil pressures and resultant forces used in this check are presented in Table 3-28.

Groundwater pressure against safety related structures is relieved by an underdrain system as described in Section 2.4.13.5.

The following Category I Structures are not founded on continuous rock but are supported on earth or weathered rock. The maximum allowable bearing pressure assumed for foundation design for structures is 3000psf for earth and 15000psf for weathered rock.

NSW + SNSW Intake Structure

NSW + SNSW Discharge Structure

NSW Electrical Conduit Manholes

Pipe Trench to Reactor Make-up and Refueling Water Storage Tanks

For each of the structures listed above, it has been determined that the structure moves with the ground motion during an earthquake to account for the soil-structure interaction effect. The seismic analysis for lateral soil-structure interaction for these facilities was performed using Mononobe-Okabe seismic coefficient analysis as described above. Also it is concluded in Section 2.5.4.8.1 that structures founded on soil and/or partially weathered rock will not undergo liquefaction or excessive deformations during the SSE.

3.7.2.5 Development of Floor Response Spectra

Figure 2-110 reflects the time-history spectra and site design spectra.

The synthetic earthquakes used to generate the time-history spectra in Figure 2-110 are used to generate response spectra at elevations in structures that house systems and components which are required to be designed for seismic excitation.

The analytical technique used to generate the response spectra at specified elevations in a structure is the time-history method or the direct generation method. The direct generation method of generating floor response spectra has been approved for use for the snubber reduction program. When using this method, the composite ground response spectra must be used as the input demand curve. When using 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. When using the direct generation method, a power spectral density function (PSD) is generated for the site ground response spectra and then used to generate a structurally amplified PSD at the required elevation.

The PSD is then converted to a response spectrum reflecting the maximum acceleration of a single degree of freedom system for a range of frequencies at the respective elevation.

Vertical response spectra are not generated. The floor slabs in Category 1 structures were examined for flexibility and were found to be sufficiently stiff to justify the assumption of negligible amplification in the vertical direction.

Damping values for the structural model are selected from Section 3.7.1.3.

TIME-HISTORY ANALYSIS

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 $a_{r\,n}$ for any arbitrary mass point r, in the nth mode, can be represented as a function of the modal displacement A_n, therefore,

$$a_{rn} = A_n \frac{a_{rn}}{A_n} = A_n \phi_{rn}$$

$$\dot{a}_{rn} = \dot{A}_n \phi_{rn}$$

$$\ddot{a}_{rn} = \ddot{A}_n \phi_{rn}$$

where,

$a_{r\,n}$ = Displacement of the rth mass point in the nth mode

$\phi_{r\,n}$ = Mode shape magnitude at mass point r, for the nth 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 nth mode this equation is:

$$\ddot{A}_n + W_n{}^2 A_n + 2B_n A_n = -\ddot{Y}_s(t)\gamma_n$$

where:

W_n = Natural circular frequency of the nth mode

B_n = $\lambda_n W_n$

λ_n = Ratio of damping to critical damping for the nth mode

$\ddot{Y}_s(t)$ = Support time-acceleration history

γ_n = Modal participation factor for the nth mode

$$\gamma n = \frac{\sum_{r=1}^{j} M_r \phi_{rn}}{\sum_{r=1}^{j} M_r \phi^2_{\ rn}}$$

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.

DIRECT GENERATION

The computer code used to perform the direct generation is called Equipment Dynamic Analysis Package (EDASP). This computer code is based on the following method of direct generation. This method is explained in greater detail by Unruh and Kana (33).

A method of direct transformation of a power density spectrum to a response spectrum without using a time-history is presented by Singh and Chu (26) with further discussions by Singh (27). Additional development of the inverse transformation is provided by Kaul (28).

The equation of motion of a single degree of freedom system which is excited at its base by an input acceleration time-history x(t) is;

$$m\ddot{y} + c(\dot{y} - \dot{x}) + k(y - x) = 0 \qquad \text{Equation (1)}$$

m = mass of the system
c = the damping coefficient
k = the system stiffness
x = the base input motion
y = the absolute response of the mass
Dots indicate differentiation with respect to time.

In terms of the relative response z = y-x

$$m\ddot{z} + c\dot{z} + kz = -m\ddot{x} \qquad \text{Equation (2)}$$

When the above equations are divided by m and $w_o = \sqrt{k/m}$ and c = $2mw_o\ \beta$ the equations become;

$$\ddot{y} + 2\beta w_o(\dot{y} - \dot{x}) + w_o^2(y - x) = 0$$ Equation (1a)

and

$$\ddot{z} + 2\beta w_o \dot{z} + w_o^2 z = -\ddot{x}$$ Equation (2a)

Integration of Equation (2a) will give the response of the system to the base input time history x. A report by Nigam and Jennings (29) details the numerical procedures necessary to obtain the peak response of the system for strong-motion earthquake events.

It is assumed that the seismic event is a stationary Gaussian random process, therefore specification of its mean and standard deviation completely describe the event. The seismic event has a zero mean and its standard deviation is obtained from its power spectral density (PSD) ϕ(w). The PSD of the system response ψ(w) is related to its base input PSD, ϕ(w) via the system transfer function in the equation which follows:

$$\psi(w, w_o) = H(W, w_o)\, x^*(w, w_o)\, \phi(w)$$ Equation (3)

where $H(w,w_o)$ is obtained from Equation (1a), via a Fourier Transfer, as

$$H(w, w_o) = \frac{Y(w)}{X(w)} = \frac{w_o^2 + i2\, w_o \beta w}{(w_o^2 - w^2) + i2 w_o \beta w}$$ Equation (3)

and $H^*(w,w_o)$ is the complex conjugate transfer function. Thus,

$$\psi(w, w_o) = \left[\frac{w_o^4 + 4w^2\beta^2 w^2}{(w_o^2 - w^2)^2 + 4w_o^2\beta^2 w^2}\right]\phi w$$ Equation (4)

The standard deviation of the response of the system is obtained from;

$$\sigma^2(w_o) = \int_{-\infty}^{+\infty} \psi(w, w_o)dw$$ Equation (5)

We define the response spectrum in terms of the standard deviation of the response of the system as;

$$R(w_o) = F(w_o) \bullet \sigma(w_o)$$ Equation (6)

F_o = Amplitude factor

The standard deviation must be multiplied by the amplitude factor to account for the peak response. When there is a small probability that the response spectrum value will be exceeded, Amin and Gunger (30) show that;

$$F_o(w_o) = \left[-2\ell n\,\{(-\pi / T)(\sigma / \sigma)\ell n(1 - r)\}\right]^{1/2}$$ Equation (7)

T = Earthquake effective time duration

sigma = standard deviation of the time derivative of the response;

$$\dot{\sigma}^2(w_o) = \int_{-\infty}^{+\infty} w^2 \psi(w, w_o) dw \qquad \text{Equation (8)}$$

r = probability of expectance

The transformation of a response spectrum into a PSD uses a method developed by Kaul (28);

$$\hat{\phi}(w_o) = \frac{\frac{2\beta}{\pi w_o} R^2(w_o)}{\{-2\ell n[-(\frac{\pi}{w_o T})\ell n(1-r)]\}} \qquad \text{Equation (9)}$$

and an iterative process. A flow chart of the iterative scheme is given in Figure 3-300. The results from equation 10 are accurate in the frequency range of 0.25 to 6.0 Hz. and the results are conservative for frequencies outside of this range for response spectra given in USNRC Regulatory Guide 1.60 (31).

This method produces inaccurate results in systems with low dampings, therefore corrections must be made to generate acceptable results. Rosenblueth and Elorduy (32) have developed a formula that corrects the system dampings;

$$\beta_e = \beta + 2/(w_o T)$$

The damping correction term is applied to all of the above expressions. The approximate solution given in Equation (10) is used as an initial estimate in the iterative process with the exact results given in Equation (7).

To illustrate the justification for the direct generation method a comparison is made between the direct generation method and the time history method. To provide the comparison a 0.5% damped ground response spectrum is generated from an artificial time history. This ground response spectrum is then used as input to the direct generation method. The same time history used to generate the ground response spectrum is also used to generate floor response spectra using the time history method. The structure used for the comparison is the reactor building interior structure. The same seismic analysis model is used for both methods. Floor response spectra are generated at 0.5% and 5% critical damping at the following mass points:

Mass Point	Elevation
3	562+0
7	595+4
11	628+8
15	662+0
19	691+2
21	713+1

The comparison of responses found in Table 3-108 contain information as follows:

Frequency columns: The EDASP column gives the frequency at which the acceleration under the heading TH-EDAS is computed.

	The T.H. column lists frequencies near those used by EDASP for which responses were computed by the original analysis.
Acceleration Columns:	The T.H. column gives the acceleration for frequency from the original seismic analysis of the reactor building.
	The TH-EDAS column gives the acceleration for the frequency from the EDASP program for the site ground response spectrum plotted from the time-history.

Comparison of accelerations (peak and ZPA) for spectra generated by time-histories and EDASP using the site ground response spectrum generated from the time-histories:

Elevation	% Peak Change		% Change at 20 hz	
	0.5%	5%	0.5%	5%
562+0	+1.8	NA	+4.5	NA
595+4	+5.1	+13.0	+30.0	+6.9
628+8	+7.6	+15.8	+23.8	+2.4
662+0	+9.4	+17.6	+5.7	0.0
691+2	+10.4	+17.6	+3.2	+3.2
713+1	+10.8	+17.5	+11.6	+4.3

The curves generated by EDASP show results were conservative at the peaks particularly for the 0.5% curves. There is more variation away from the peak, but EDASP produces conservative results. EDASP provides acceptable elevated response spectra for seismic design. Use of the direct generation method is also consistent with the recommendation in NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria".

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 spectrum. The numerical average for the response of the four earthquake time-histories was used to generate the final response spectrum used in the seismic design.

A typical structural mathematical model of the Containment Interior Structure is shown in Figure 3-217.

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 six elevations of the Containment interior structure are shown in Figure 3-235 through Figure 3-240.

TORSIONAL CONTRIBUTION TO FLOOR RESPONSE SPECTRA

When the torsional contribution is significant, as in the case of the Auxiliary Building, floor response spectra are generated at various points of interest of each floor elevation for the horizontal components of earthquake motion. Extra weightless joints, connected to the center of structural rigidity of their respective floor elevations by horizontal members of relatively large stiffness, are added to the three dimensional model to generate spectra at locations near the periphery of the structure.

3.7.2.6 Three Components of Earthquake Motion

1. Structures

   The earthquake ground motions are assumed to act in one of two perpendicular horizontal directions simultaneously with motion in the vertical direction. The structure is then designed for the case of vertical and horizontal earthquake motion giving the more severe stresses. The provisions of Regulatory Guide 1.92 are not applicable to the design of the Catawba Nuclear Station structures due to the implementation date of the guide.

   A comparison was made for the containment interior structure to determine the effects of going to a three dimensional design philosophy. The original lumped mass model of the interior structure was reanalyzed. The new analysis accounts for all the provisions now found in Regulatory Guide 1.92.

   A comparison of design forces for the new three dimensional analysis versus the original two dimensional analysis is provided in Table 3-27.

   The shear values (Sy and Sz) in Table 3-27 show less than 1% change as a result of including the three dimensional effects. The axial loads (Ax) range from conservative results at the base to results that are not conservative in the upper portions of the structure. When compared to the capacity of the structural members, the increases in axial loads are negligible. Consideration of the effects of the three dimensional earthquake add little to the overall design and decrease the overall structural integrity by negligible amounts.

   While Regulatory Guide 1.92 is not applicable to Catawba structures, the above comparison demonstrates that the Catawba design meets the intent of this guide.

2. Systems and Components (Westinghouse)

   The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equipment, structures, and components. The Westinghouse system and equipment response is determined using three earthquake components, two horizontal and one vertical. The design ground response spectra are the bases for generating these three input components. The damping values used in the analysis are those given in Section 3.7.1.3.

   In computing the Westinghouse system and equipment response by response spectrum modal analysis, the methods of Section 3.7.2.7 are used to combine all significant modal responses to obtain the combined unidirectional responses.

   For each horizontal direction of shock, the total response is obtained by taking the absolute sum of the combined unidirectional responses for the horizontal and vertical directions. The most conservative results, with each direction of horizontal shock considered, are used to obtain the critical response values.

3.7.2.7 Combination of Modal Responses

The overall structural response for Duke designed structures is obtained by combining the modal contributions of all the modes considered. This is accomplished using the square root of the sum of the squares as discussed in Section 3.7.2.1. The provisions of Regulatory Guide 1.92 are not applicable to the design of the Catawba Nuclear Station structures.

3.7.2.7.1 Westinghouse Scope

For analysis under the Westinghouse scope, the total unidirectional seismic response is obtained by combining the individual modal responses utilizing the square root of the sum of the squares 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 percent of the lower frequency. Groups are formed starting from the lowest frequency and working towards successively higher frequencies. No one frequency is included in more than one group. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor ε. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{\mathrm{k=mj}}^{N_j - 1} \sum_{\ell=k+1}^{N_j} R_k R_\ell \varepsilon_{K\ell} \qquad \text{Equation 3.1}$$

Where:

R_T = total unidirectional 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 models

N_j = highest modal number associated with group j of closely spaced models

$\varepsilon_{\mathrm{Kliter}}$ = coupling factor with

$$\varepsilon_{k\ell} = \left[1 + \left(\frac{\omega'_k - \omega'_\ell}{(\beta'_k \omega_k + \beta'_\ell \omega_\ell} \right)^2 \right]^{-1}$$

and:

$$\omega'_k = \omega_k \left[1 - (\beta_k)^2\right]^{1/2}$$

$$\beta'_k = \beta_k + \frac{2}{\omega_k t_d}$$

where

ω_K = frequency of closely spaced mode K

β_K = fraction of critical damping in closely spaced mode K

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
Frequency	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 [3.1];

$$R_T^2 = \left[R_1^2 + R_2^2 + R_3^2 + \ldots + R_8^2\right] + 2 R_2 R_3 \ \varepsilon_{23} + 2 R_2 R_4 \varepsilon_{24} + 2R_3 R_4 \varepsilon_{34} + 2 R_7 \varepsilon_{67}$$

The worst condition for closely spaced modes, i.e., largest number in the lowest frequencies, is the Auxiliary Building. Subsequent analysis indicates that the greatest increase in response, due to consideration of these closely spaced modes, is 7.2%, with an average increase of 2.2%. This increase in seismic responses due to closely spaced modes is insignificant when compared with the original seismic responses.

3.7.2.7.2 Unit 1 NSSS Reanalysis for Steam Generator Replacement (Unit 1)

Modal responses for the Unit 1 NSSS, which includes Babcock and Wilcox steam generators, are combined in accordance with Regulatory Guide 1.92 as described in Section 3.7.3.7.

3.7.2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

Complete separation of Seismic Category I structures from adjacent Seismic Category I or non-Category I structures prevents interaction with adjacent structures. The non-Category I structures with sufficient mass to possibly impair the integrity of Seismic Category I structures or components upon collapse are analyzed to prevent their failure in the direction of a Seismic Category I structure under SSE conditions in a manner such that the margin of safety of these structures in equivalent to that of Seismic Category I structures. The collapse of any non-

Category I structure, not analyzed to prevent their failure under SSE conditions, will not impair the integrity of Seismic Category I structures or components.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The provisions of Regulatory Guide 1.122 are not applicable to the design of the Catawba structures due to the Regulatory Guide implementation date; however, its intent is met.

To account for possible variations in structural properties, bedrock properties and damping, the calculated floor response spectra are enveloped to provide a ±10 percent shift in period and an additional 10 percent increase in peak response.

The 10 percent peak widening utilized is less than the provisions of Regulatory Guide 1.122, however, a major contributory factor in the Regulatory Guide is soil-structure interaction, which is not applicable to the major Catawba Category I Structures which are founded on sound continuous rock. In addition, the amplitude increase mentioned above is not included in Regulatory Guide 1.122 and, thus, is more conservative.

An adjusted design envelope of a typical floor response spectrum utilizing the 10 percent period shift is shown in Figure 3-240.

An alternative method for the broadening of the structural peaks for use in seismic piping analysis is an envelope of the response of the piping system to shifted floor response spectra. This method is described in the Code Case N-397 and the Summer 1984 Addendum to Section III, Appendix N of the ASME Boiler and Pressure Vessel Code.

3.7.2.10 Use of Constant Vertical Static 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 20Hz 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.

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 3.7.2.1.2.

Constant vertical static factors are not used as the vertical floor response load for the seismic design of safety classed systems and components within Westinghouse's scope of responsibility. All such systems and components are analyzed in the vertical directions.

3.7.2.11 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 are computed. The shears due to torsional moments are applied to the frames by the relative stiffness method as presented in Reference 3, "Design of Multistory Reinforced Concrete Buildings for Earthquake

Motions," by Blume, Newmark, and Corning. A mathematical model with a rigid link connecting the center of mass and center of rigidity at each floor or support elevation is used to calculate the actual torsional responses. For a detailed description of the formulation of the remainder of the model and the dynamic analysis, refer to Section 3.7.2.1.1.

Refer to Section 3.7.2.5 for a description of how the effects of building torsion are included in the floor response spectra.

3.7.2.12 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 do not necessarily coincide.

Table 3-26 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 2-110 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 greater than 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.

3.7.2.13 Methods for Seismic Analysis of Dams

The methods for the seismic analysis of dams are defined in Section 2.5.6.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

The overturning moments and base shears due to seismic forces for Category I structures are determined using the response spectrum technique. A full description of this method is given in Section 3.7.2.1. Figure 3-233 and Figure 3-234 give examples of these forces for the Reactor Building interior structure.

The overturning moments for shell type structures, e.g., Reactor Building and Containment vessel, are automatically included in the shell analysis of such structures.

3.7.2.15 Analysis Procedure for Damping

Refer to Section 3.7.1.3 for damping values used for Category I structures. The assumptions on which these values are based are given in the same Section.

No coupled systems are used in the structural seismic analysis. For the damping values used for components and systems refer to Section 3.7.3.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

3.7.3.1.1 Piping

Seismic analysis methods applicable to piping are described in Section 3.7.3.8. As an alternative to the method described in Section 3.7.3.8, the independent support motion (ISM) methodology may be used as described below.

3.7.3.1.1.1 Independent Support Motion Methodology

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 34) and was approved for use by NRC's letter of October 13, 1995 (Reference 35).

3.7.3.1.2 Westinghouse Scope

Dynamic analyses applicable to subsystems within Westinghouse's scope of responsibility are, in general, performed using a modal analysis plus either the response spectrum analysis or integration of the uncoupled modal equations, or by direct integration of the coupled differential equations of motion.

Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for mathematical analysis. The essence of this step is to select a model 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. Some typical modeling techniques are presented in Reference 12.

Equations of Motion

Consider the multi-degree of freedom system shown in Figure 3-242. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \frac{\mathrm{i}}{\Sigma} c_{ri} \dot{\mathrm{u}}_i + \frac{\mathrm{i}}{\Sigma} k_{ri} u_i = 0 \qquad \text{Equation [3.2]}$$

where,

m_r = the value of the mass moment of rotational inertia at mass point r

y_r = absolute translational or angular acceleration of mass point r

c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity

$\dot{u}_i$ = translational or 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 positive displacement (rotation)

u_i = displacement (rotation) of mass point i relative to the base

As an example, note that Figure 3-242 does not attempt to show all of the springs (and none of the dashpots) which are represented in Equation [3.2].

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad \text{Equation [3.3]}$$

where

$\ddot{y}_s$ = absolute translational (angular) acceleration of the base

$\ddot{u}_r$ = translational (angular) acceleration of mass point r relative to the base

Equation [3.2] can be written as:

$$m_r\ddot{u}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri}u_i = -m_r\ddot{y}_s \qquad \text{Equation [3.4]}$$

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

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad \text{Equation [3.5]}$$

<u>Modal Analysis</u>

<u>Natural Frequencies and Mode Shapes</u>

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation [3.4]. The right hand side and the damping term are set equal to zero for this purpose as illustrated in Reference 11. Thus, Equation [3.4] becomes:

$$m_r\ddot{u}_r + \frac{i}{\sum}k_{ri}u_i = 0 \qquad \text{Equation [3.6]}$$

The equation given for each mass point r in Equation [3.6] can be written as a system of equations in matrix form as:

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

where

[M] = mass and rotational inertia matrix

$\{\Delta\}$ = 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 general translational and angular accelerations at each mass point relative to the base, $d^2\{\Delta\}/dt^2$

Harmonic motion is assumed and the $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\}\sin\omega t \qquad \text{Equation [3.8]}$$

where

$\{\delta\}$ = column matrix of the spatial displacement and rotation at each mass point relative to the base

ω = natural frequency of harmonic motion in radians per second

The displacement function and its second derivative are substituted into Equation [3.7] and yield:

$$[K]\{\delta\} = \omega^2[M]\{\delta\} \qquad \text{Equation [3.9]}$$

The determinant $|[K] - \omega^2[M]|$ set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation [3.9] This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthoganol to each other and are sometimes referred to as normal mode vibrations. 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:

$$k - \omega^2 m = 0 \qquad \text{Equation [3.10]}$$

or:

$$\omega = \sqrt{k/m}$$

where ω is the natural angular frequency in radians per second.

The natural frequency in cycles per second is therefore:

$$f = \frac{1}{2\pi}\sqrt{k/m} \qquad \text{Equation [3.11]}$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation [3.9].

Modal Equations

The response of a structure or component is always some combination of its normal modes. 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 11):

$$\ddot{A}_n + 2\omega_n p_n \dot{A}_n + \omega_n^2 A_n = -r_n \ddot{y}_s \qquad \text{Equation [3.12]}$$

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

$$u_{r\,n} = A_n \phi_{r\,n} \qquad \text{Equation [3.13]}$$

where

ω_n = natural frequency of mode n in radians per second

p_n = critical damping ratio of mode n

r_n = modal participation factor of mode n given by:

$$r_n = \frac{\sum^{n} m_r \phi^1_{r\,n}}{\sum^{n} m_r \phi^2{}_{r\,n}} \qquad \text{Equation [3.14]}$$

and

$\phi^1_{r\,n}$ = value of $\Phi_{r\,n}$ in the direction of the earthquake

The essence of the modal analysis lies in the fact that Equation [3.12] is analogous to the equation of motion for a single degree of freedom system that will be developed from Equation [3.5]. Dividing Equation [3.5] by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad \text{Equation [3.15]}$$

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

$$p = \frac{c}{c_c} \qquad \text{Equation [3.16]}$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \qquad \text{Equation [3.17]}$$

Substituting Equation [3.17] into Equation [3.16] and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad \text{Equation [3.18]}$$

Substituting this expression and the expression for k/m given by Equation [3.10] into Equation [3.15] gives:

$$\ddot{u} + 2\omega p\dot{u} + \omega^2 u = -\ddot{y}_s \qquad \text{Equation [3.19]}$$

Note the similarity of Equations [3.12] and [3.19]. 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. There are three ways used to overcome this limitation when considering a slightly damped structure (e.g., steel) supported by a massive moderately damped structure (e.g., concrete).

The first method is to develop and analyze separate mathematical 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 is used as a forcing function for the subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion. The third method is to use the Rayleigh damping method based on computed modal energy distribution.

Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Reference 12) (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.

The response spectrum concept can be best 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 is calculated for a given base motion. The variations in response are established and the maximum absolute 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. The spectral acceleration, velocity, and displacement are related by the equation:

$$^{S}a_n = \omega_n \, ^{S}v_n = \mathrm{w}_n^2 \, \mathrm{S}_{dn} \qquad \text{Equation [3.20]}$$

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 ground response spectrum or ground time history 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. These are called floor response spectra.

Integration of Modal Equations

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

Integration procedure for the uncoupled modal Equation [3.12] to obtain the modal displacements and acce

1. lerations as a function of time.
2. Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

Integration Procedure

Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure 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.

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 [3.13] as:

$$u_{rn} = A_n \phi_{rn} \qquad \text{Equation [3.21]}$$

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

$$\ddot{u}_{rn} = \ddot{A}_n \phi_{rn} \qquad \text{Equation [3.22]}$$

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

Integration of Coupled Equations of Motion

The dynamic transient analysis is a time history solution of the response of a given structure to known forces and/or displacement forcing functions. The structure may include linear or nonlinear elements, gaps, interfaces, plastic elements, and viscous and Coulomb dampers. Nodal displacements, nodal forces, pressure, and/or temperatures may be considered as forcing functions. Nodal displacements and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\ \{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \qquad \text{Equation [3.23]}$$

where the terms are as defined earlier and {F(t)} may include the effects of applied displacements, forces, pressures, temperatures, or nonlinear effects such as plasticity and dynamic elements with gaps. Options of translational accelerations input to a structural system and the inclusion of static deformation and/or preload may be considered in the nonlinear

dynamic transient analysis. The option of translational input such as uniform base motion to a structural system is considered by introducing an inertia force term of $[M]\{\ddot{z}\}$ to the right hand side of the basic Equation [3.23], i.e.,

$$[M]\{\ddot{x}\}+[C]\{\dot{x}\}+[K]\{x\}=\{F\}-[M]\{\ddot{z}\} \qquad \text{Equation [3.24]}$$

The vector $\{\ddot{z}\}$ is defined by its components $\{\ddot{z}_i\}$ where i refers to each degree of freedom of the system. $\ddot{z}_i$ is equal to a_1, a_2, or a_3 if the i-th degree of freedom is aligned with the direction of the system translational acceleration a_1, a_2, or a_3, respectively. $\ddot{z}_i = 0$ if the i-th degree of freedom is not aligned with any direction of the system translational acceleration. Typical application of this option is a structural system subjected to a seismic excitation of a given ground acceleration record. The displacement {x} obtained from the solution of Equation [3.24] is the displacement relative to the ground.

The option of the inclusion of initial static deformation or preload in a nonlinear transient dynamic structural analysis is considered by solving the static problem prior to the dynamic analysis. At each stage of integration in transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the external forces which are statically applied. Hence, only the portion of the forces which deviate from the static loads will produce dynamic effects. The output of this analysis is the total result due to static and dynamic applied loads.

One available method for the numerical integration of Equations [3.23] and [3.24] is the Newmark Beta integration scheme proposed by Chan, Cox, and Benfield (Reference 13). In this integration scheme, Equations [3.23] and [3.24] are replaced by:

$$\begin{aligned} &\frac{1}{(\Delta t)^2}[M]\{x_{n+2}-2x_{n+1}+x_n\}+\frac{1}{2(\Delta t)}\{x_{n+2}-x_n\}[C] \\ &1/3+[K]\{\beta x_{n+2}+(1-2\beta)x_{n+1}+\beta x_n\} \\ &\quad=\{\beta F_{n+2}+\ (1-2\beta)F_{n+1}+\beta F_n\} \end{aligned} \qquad \text{Equation [3.25]}$$

where

n, n+1, n+2	=	past, present, and future (updated) values of the variables
β	=	parameter to be selected on the basis of numerical stability and accuracy
F	=	the total right hand side of the equation of motion (Equation [3.23] or [3.24])
Δt	=	$t_{n+2} - t_{n+1} = t_{n+1} - t_n$

The value of β is chosen equal to 1/3 in order to provide a margin of numerical stability for nonlinear problems. Since the numerical stability of Equation [3.25] is mostly determined by the left hand side terms of that equation, the right hand side terms were replaced by F_{n+1}. Furthermore, since the time increment may vary between two successive time substeps, Equation [3.25] may be modified as follows:

$$\frac{2}{(\Delta t+\Delta t_1)}[M]\{\frac{x_{n+2}-x_{n+1}}{\Delta t_1}\ -\ \frac{x_{n+1}-x_n}{\Delta t}\}+\frac{1}{\Delta t+\Delta t_1}[C]\{x_{n+2}-x_n\} \qquad \text{Equation [3.26]}$$

$+1/3[K] \{x_{n+2} + x_{n+1} + x_n\} = \{F_{n+1}\}$

By factoring x_{n+2}, x_{n+1}, and x_n, and rearranging terms, Equation [3.26] obtained as follows:

$$\{C_5 [M] + C_3[C] + (1/3) [K]\} \{x_{n+2}\} = \{F_{n+1}\}$$
$$+ \{C7 [M] – (1/3)[K]\} \{x_{n+1}\} \qquad \text{Equation [3.27]}$$
$$+ \{-C_2 [M] + C_3[C] – (1/3)[K]\} \{x_n\}$$

where

$$C_2 = \frac{2}{\Delta t_1(\Delta t + \Delta t_1)}$$

$$C_3 = \frac{1}{\Delta t + \Delta t_1}$$

$$C_5 = \frac{2}{\Delta t(\Delta t + \Delta t_1)}$$

$$C_7 = C_2 + C_5$$

The above set of simultaneous linear equations is solved to obtain the present values of nodal displacements $\{x_t\}$ in terms of the previous (known) values of the nodal displacements. Since [M], [C], and [K] are included in the equation, they can also be time or displacement dependent.

3.7.3.2 Determination of Number of Earthquake Cycles

1. NSSS System

   Where fatigue analyses of mechanical systems and components within Westinghouse scope are required, Westinghouse specifies in the equipment specification the number of cycles of the OBE to be considered. The number of cycles for NSSS components is given in Table 3-50.

2. ASME, Section III, Class I piping other than NSSS

   For the design of Class I piping, an average of 5 equivalent operational basis earthquakes (OBE) and a total of 200 stress cycles for piping systems will be used for the full plant lifetime. One safe shutdown earthquake (SSE) with 100 stress cycles for piping systems will be used.

3.7.3.3 Procedure Used for Modeling

Refer to Section 3.7.3.1.2 for modeling procedures for subsystems in Westinghouse scope of responsibility.

Seismic piping other than the NSSS is analyzed as a number of seismic subsystems. The response of the supporting structure (a seismic system) is an input to these analyses.

3.7.3.4 Basis for Selection of Frequencies

In theory, the seismic response of piping can be reduced by designing it to have a fundamental frequency much different from that of the supporting structure. In application, the range of practical piping frequencies is limited by other factors. Too flexible a system can have

excessive sag, weight stresses, and vibration during normal operation; too rigid a system results in a congested and costly array of supports, particularly where thermal expansion is present. For these reasons, the piping typically has some dynamic modes at high - response forcing frequencies of the structure. The piping analysis methods described in Section 3.7.3.8 account for this, and the piping is designed to withstand the resulting loads.

The analysis of equipment subjected to seismic loading in the Westinghouse scope 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 equipment will show very little response.
3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

In all cases, equipment under earthquake loadings is designed to be within code allowable stresses.

Also, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

3.7.3.5.1 For NSSS

The equivalent static load 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 systems 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. If the equivalent static load method is used for seismic Category I piping, an example will be provided to the NRC.

3.7.3.5.2 Other Than NSSS (Seismic Category I and II Piping only)

The equivalent static load method may be used provided it has been demonstrated that the subsystem is either rigid, or can be adequately and realistically represented as a single degree of freedom system. In this case, the modal participation factor is considered to be one and the equivalent static load is determined by using the peak acceleration of the applicable floor response spectrum at the appropriate value of damping.

The method may also be used, when the dynamic characteristics of the subsystem has not been determined, provided the subsystem can be realistically represented as a simple model.

In this case a factor of 1.5 is applied to the peak acceleration of the applicable floor spectrum at the appropriate value of damping. If the equivalent static load method is used for seismic Category I piping, an example will be provided to the NRC.

The relative motion between support points is accounted for in the simplified analysis by using the method described in Section 3.7.3.9.

3.7.3.6 Three Components of Earthquake Motion

Methods used to account for three components of earthquake motion for Westinghouse subsystems are given in Section 3.7.2.6.

For seismic piping other than NSSS, analysis is performed using simultaneous three-direction excitation. Directional responses are combined into a total response by taking the square root of the sum of the squares (SRSS) of individual responses. This method conforms fully to the recommendations of Regulatory Guide 1.92.

3.7.3.7 Combination of Modal Responses

Methods used to combine modal responses for subsystems in Westinghouse's scope of responsibility are given in Section 3.7.2.7.

For seismic piping other than the NSSS, modal responses are combined into a total response by taking the square root of the sum of the squares (SRSS) of individual responses. The responses from groups of closely spaced modes, defined as having frequencies between the lowest frequency in the group and a frequency ten percent higher, are combined by absolute summation; the resulting response for each group is then combined by SRSS with the remaining responses from the modes which are not closely spaced. This method conforms fully to the recommendations of Regulatory Guide 1.92.

3.7.3.8 Analytical Procedures for Piping

The criteria for determining which piping is to be seismic are discussed in Section 3.2.1. All safety-related piping is classified as Seismic Category I. The specific analytical procedures used in qualifying a pipe depend on its size, temperature, structural frequency, and other factors as discussed in this section.

Accuracy of design documents used as inputs to seismic piping analyses was identified as an NRC concern in NRC IE Bulletin 79-14, issued July 2, 1979. Assurance that design documents reflect as-built conditions is detailed in Duke's response to IE Bulletin 79-14 (Reference 43). The quality assurance program at Catawba fully addresses those conerns identified by IE Bulletin 79-14. Fundamental throughout this program are numerous checks to confirm that the seismic analysis of Seismic Category I (i.e., Duke QA Condition 1) piping systems accurately reflect the as-built conditions which exist. Duke Power's rigorous compliance with the quality assurance program at Catawba assures that the requirements of IE Bulletin 79-14 have been and will continue to be satisfied.

3.7.3.8.1 Static Analysis of Rigid Piping

Piping subsystems with a period of less than 0.0303 seconds are considered rigid. This piping is designed for a uniform static coefficient equal to the maximum floor acceleration at the appropriate location in the structure.

3.7.3.8.2 Rigorous Analysis of Flexible Piping

Piping subsystems with a period greater than 0.0303 seconds are considered flexible. Some of this piping can be handled by the simplified, conservative alternate analysis described in Section 3.7.3.8.3. The remaining flexible pipe is analyzed using the modal response spectrum method, as follows:

Each pipe 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. Typically, all modes having a period greater than 0.0303 seconds are used in the analysis. In cases where the seismic model for a particular pipe is very large, a lesser number of modes may be used, provided the omitted modes lie in the flat region (rigid side) of the applicable response spectrum. This assures that the results include all significant contributions.

The mode shapes and frequencies are solved in accordance with the following equation:

$$(K - w_n^2 M)\,\phi n = 0$$

in which:

K = square stiffness matrix of the pipe loop

M = mass matrix for the pipe loop

w_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 mode is found by solving the following equation:

$$Y_n \max = \frac{R_n Sa_n D}{M_n w_n^2}$$

in which:

$Y_n$max = maximum characteristic response of the n^th mode

Rn = mass participation factor for the n^th mode = $\sum M_i \phi_{in}$

Sa_n = spectral acceleration for the n^th mode

D = earthquake direction coefficient

M_n = generalized mass matrix for the n^th mode = $\sum M_i \phi_{in}^2$

M_i = Mass of the i^{th} mass

ϕ_{in} = Modal displacement of the i^{th} mass for the n^{th} mode

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

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

$$Q_n = KV$$

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, in a manner similar to that done for displacements. Combination methods for the three spatial components and treatment of closely-spaced modes are discussed in Sections 3.7.3.6 and 3.7.3.7.

The results of the rigorous dynamic analysis include seismic deflections and valve accelerations, as well as the seismic contributions to pipe stress, equipment nozzle loads, and pipe support/restraint loads. These are combined with other load cases required to qualify the piping and components, as discussed in Section 3.9.3.

3.7.3.8.3 Alternate Analysis of Flexible Piping

Seismic piping subsystems which meet all of the requirements of Table 3-29 fall under the scope of Alternate Analysis, and may be analyzed using the following simplified procedure:

A rigorous spectral analysis is employed to derive the required pipe design data for earthquake loading. In this analysis, the effects associated with deadweight of the pipe are coupled with the seismic effects and evaluated simultaneously. In view of the exceedingly large number of pipe cross-sections, weight cases and spectral cases involved, it would be impractical to analyze every pipe situation. A simplification in the analytical effort is achieved for a given pipe model and spectra by limiting all pipe cross-sections, regardless of weights, to the same maximum dynamic deflection. This is equivalent to specifying the same unique set of modal frequencies

and accelerations for all pipe cases. Due to this specification, a spectral analysis of any one pipe cross-section would be sufficient for determining pipe stresses and reactions for all other pipe cross-sections as these would be interrelated for any two cross-sections by proportionality constants. It should be emphasized that this method is identical to performing a complete modal analysis for each pipe case and no accuracy is sacrificed.

The above method does result in varying degrees of conservatism for different pipe cases if a single maximum dynamic deflection is specified for all pipe cases regardless of weight, cross-section properties or response spectra. This drawback has been removed by separating the pipe cases into seven groups dependent on spectra and weight. A single critical cross-section is selected for each of the seven groups and studied by analyzing various configurations (with changes in direction) that are likely to occur. Support spacing, support loads, and nozzle loads are determined from these studies and used in the above method to obtain results for other cross-sections within a group. Unusually high support loads, e.g. loads on axial restraints, are not included in this method and are studied separately. The increased support loads and reduced spans associated with concentrated weights, elbow, tees, etc. are also studied separately.

A displacement criterion allowing maximum OBE and SSE displacements of one inch was imposed on the analysis. This arbitrary criterion was adopted to avoid the possibility of excessive displacements.

Evaluation of Seismic Anchor Motion (SAM) analysis may be made through the use of Anchor Movement tables. Assumptions made regarding the effects of the SAM displacements upon the piping being analyzed are that all floors in a single structure move in-phase, and, when piping attaches to two structures, the structures move out of phase.

No separate evaluation is made for the stress requirement of the Faulted Condition because it is covered by the stress requirement of the Upset Condition if only pressure, gravity and earthquake loadings are considered as they were in the subject analysis. For the Faulted Condition only the seismic (SSE) contribution to the total stress is increased over the seismic (OBE) contribution to the total stress for the Upset Condition with the gravity and pressure stresses being equivalent for both design conditions. The stress (displacement, reaction) effect of a SSE is taken to be 15/8 that of the corresponding OBE. Since the allowable stress for the Faulted Condition is double that of the Upset Condition, the evaluation of earthquake stress limit is based on the Upset Condition only.

Spans with concentrated weights are evaluated against the displacement and stress criteria described above; in addition, acceleration limits are imposed assuming that concentrated weights are usually valves. The limits are 2 g (SSE) in the vertical direction and 3 g (SSE) in each of two mutually perpendicular horizontal directions. All three of these acceleration components may occur simultaneously, but resultant combinations of these components are not used as acceleration limits. The limits are independent of the orientation of a valve.

3.7.3.8.4 Response to NRC IE Bulletin 88-05 and Supplements

The NRC issued IE Bulletin 88-05, "Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey", on May 5, 1988 and Supplement 1 on June 15, 1988. The purpose of this bulletin and supplement was to alert licensees of suspect materials supplied by the above companies and request that licensees 1) take actions to assure that such materials comply with the ASME Code and design specification requirements or are suitable for their intended service, or 2) replace such materials. Upon receipt of the initial bulletin, Duke Power Company began conducting a thorough research and tracking effort to locate the subject suspect materials at Catawba

Nuclear Station. On August 3, 1988, the NRC issued IE Bulletin 88-05, Supplement 2, which directed licensees to stop further research. Duke Power Company submitted its response for IE Bulletin 88-05 and Supplements 1 and 2 in the letter from H.B. Tucker to the NRC, dated September 9, 1988. This response contained the results of the research and evaluation efforts taken up to the point when Supplement 2 to the bulletin was issued by the NRC. All identified materials in warehouse stock were removed to preclude the possibility of further use. Evaluations and necessary Justifications for Continued Operation were submitted for the remainder of the items installed on safety-related systems at Catawba Nuclear Station.

3.7.3.9 Multiply-Supported Equipment Components with Distinct Inputs

For seismic piping (other than NSSS), analysis includes earthquake loads represented by horizontal earthquake response spectra at the various floor elevations in the Category I structures. For a piping system spanning between two or more elevations (spectra), the response spectrum analysis is performed using an envelope of all appropriate floor response spectras through which the pipe passes. The spectrum used to represent the vertical seismic accelerations is two-thirds of the horizontal ground spectrum where no vertical floor spectra is developed.

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 using a static analysis. 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.

When response spectrum methods are used to evaluate Reactor Coolant System primary components interconnected between floors, the procedures of the following paragraphs are used. There are no components in Westinghouse scope of analysis which are connected between buildings. The primary components of the Reactor Coolant System are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis is the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The results of the building analysis are reviewed on a mode-by-mode basis to determine the differential motion in each mode. Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB-3650). For components, the differential motion will be evaluated as a free end displacement, since, per NB-3213.19, examples of a free end displacement are motions "that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping". The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB-3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses. All analyzed piping stresses are considered primary stresses for pipe support design.

3.7.3.10 Use of Constant Vertical Static Factors

For seismic piping subsystems, the simultaneous three-directional excitation used in the analysis does not involve constant vertical static factors.

3.7.3.11 Torsional Effects of Eccentric Masses

For seismic piping, significant masses offset from the pipe centerline are specifically included in the seismic math model. Therefore, any forces or moments, including torsion, due to these eccentric masses appear in the results of the analysis. Typical examples of such masses are remote-actuated valve operators and local bypass piping.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

The Nuclear Service Water System includes buried seismic piping connecting various safety-related structures. Due to its early position in the procurement and the erection schedules, the Nuclear Service Water System large diameter piping was purchased in accordance with the 1971 ASME Code. Procurement specifications are not design specification. This 1971 ASME Code was used for minimum pipe wall thickness determinations and for pipe stress allowable. The Nuclear Service Water system is designed and stamped in accordance with the ASME Section III Division 1 of the 1974 Edition including the Summer Addenda. The ASME Code is silent on buried piping in that it does not specify how to bury it. The full provisions of the code with respect to loading combinations and stress intensification factors were not applicable. The seismic analysis performed on this pipe considers:

1. Inertial effects due to the dynamic behavior of the material in which the pipe is embedded. The model assumes that the seismic deformations of the surrounding material are imposed on the piping, which then conforms to this deformation away from the bends and changes in direction. Friction between the pipe and soil is accounted for in analyzing pipe lengths shorter than that required for the soil to develop the full axial friction force. For buried pipe bends or changes in direction and/or penetrations, the resistance of the surrounding soil to the displacement of the pipe is accounted for using the coefficient of subgrade reaction and equations for beams or elastic foundations.

   The type of waves considered most appropriate in the analyses for buried pipeline systems are traveling body waves-compression (P) and shear (S) waves. The angles of incidence of the waves are selected to yield maximum values of strain, curvature or stress for a given condition of analysis. The wave propagation velocity with respect to the buried pipe or the ground surface is relevant for calculating the strains and curvatures induced in the buried pipeline as well as the relative displacements and rotations at the pipeline bends and junctions (References 18 and 19). The velocity of the traveling wave propagating through the entire soil/rock profile is a function of the wave length. The angle of incidence of body waves with respect to the ground surface was assumed to be in the range of 35° to 40° from vertical for short wave lengths (less than the depth to very dense soils/partially weathered rock) in order to compute a conservative lower bound velocity of propagation for analyses of the buried systems. For long wave lengths, the material velocity of the very dense soil-partially weathered rock, or a shear wave propagation velocity of 2000 ft per second, was assumed as the propagation velocity. The corresponding compression wave propagation velocity is 4200 ft/second.

Compressive or longitudinal wave effects are not excluded from consideration for seismic analyses of buried piping and electrical conduit runs, and are considered as explained in Reference 16. The pipe stresses calculated due to these compressive wave effects are less critical than those computed from the effects of shear waves.

Surface (Rayleigh) waves are not excluded from consideration for seismic analyses of buried piping and electrical conduit runs. For the Catawba site, the long period Rayleigh waves would propagate at the velocity of the bedrock, or 6000 fps for wave lengths equal to or greater than about 100 ft, twice the typical depth to rock. The pipe stresses calculated due to such Rayleigh waves are less critical than those computed from the effects of travelling body waves.

The equations of References 5, 6, and 16 are used to calculate the resulting stresses. A summary of information used in the analysis is as follows:

a. Ground motions read from ground motion spectrum shown on Figure 11 of Reference 19, after scaling to appropriate maximum acceleration.

   For each wave type, the full ground motion was assumed for calculating pipe stresses. This is conservative, since the ground motion is actually the result of summing the contribution from each of the wave types.

b. Seismic shear wave propagation velocity - 2000 fps; compression wave propagation velocity - 4200 fps.

c. Maximum ground acceleration - Bedrock, 0.15g., Backfill 0.40g.

d. Coefficient of subgrade reaction (k_o) for site backfill material (varies with pipe size).

   Average value - 1,580 psi/in

e. Friction coefficient - 0.3

f. Poisson's ratio - 0.35 for soil, 0.25 for rock

g. Shear wave velocity of rock 7000 fps, Rayleigh wave velocity 6000 fps.

Information on the other properties of the subsurface materials is presented in Chapter 2.

Deleted Paragraph Per 2001 Update.

The coefficient of subgrade reaction (k_o) was determined by the method described in Reference 16 (and outlined below) and the seismic wave material velocity obtained from the site test data. Varying the velocity will result in an increase and decrease of the coefficient; this will respectively increase and decrease stress on the pipe. For calculation purposes the velocity was assumed to vary by ±25 percent. The resulting maximum stress value thus obtained is still less than the allowable pipe stress.

As described in Reference 16, the coefficient of subgrade reaction is a function of the elastic modulus and Poisson's ratio of the soil. While field determination of the coefficient of subgrade reaction for foundation analyses is sometimes done by plate load tests, such a method for determination of the coefficient for analysis of buried pipes (subject to seismic deformations) is not appropriate and would result in an unconservative analysis due to the low value of the coefficient that would be so obtained. For use in seismic design of pipes, the coefficient of subgrade reaction should be evaluated from the elastic properties of the soil (as determined from dynamic tests) and an appropriate equation relating the variables as follows:

$$K = k_o D$$

K = soil spring constant per unit length (F/L^2)

k_o = coefficient of subgrade reaction (F/L^3)

$$k_o D = 0.65 \left[\frac{E_s D^4}{E_p I_p} \right]^{1/12} \frac{Es}{1 - V_s^2}$$

where $E_s = 2(1 + V_s) G_s$

$G_s = \rho v_s^2$

where

G_s = shear modulus of soil

ρ = mass density = Total Unit Weight ÷ Gravity Constant

v_s = shear wave velocity as determined from field tests

V_s = Poisson's ratio of soil

E_p = modulus of elasticity of the pipe material

I_p = moment of inertia of pipe about the axis of bending

D = pipe diameter

The above equation relation k_o and thus K to the properties of the soil and stiffness of the pipe (beam) is attributed to Vesic (1961).

The coefficient of subgrade reaction computed for various pipe sizes for Gs = 10,994 psi, V_s = 0.35, γ = 120 pcf, Ep = 30×10^6 psi, is tabulated as follows:

Pipe diameter, D inches	ko psi/in.
10.75	1652
12.75	1409
30	640
42	470
48	415

Comparison of the above values of coefficient of subgrade reaction related to dynamic values of soil modulus with those available in the published literature from plate load tests at larger deformations and under static loading would be inappropriate. However, for comparison purposes, it is indicated that k_o above has approximately the same meaning as $\overline{k}_{sl}$ and $\overline{k}_{hl}$ for a soil whose modulus is constant with depth in Terzaghi's notation (Terzaghi, Karl, "Evaluation of Coefficients of Subgrade Reaction.") (Geotechnique, December, 1955); $\overline{k}_{sl}$ of Terzaghi is not dependent on the flexibility of the beam (pipe) whereas k_o depends on the stiffness of the pipe relative to the soil. The coefficient k_o is indicated to have also approximately the same meaning as k' in Barkan, 1962, relating spring constants of rigid foundations under vibratory loading to the concept of elastic-subgrade reaction. Barkan provided values of the coefficient k' equal to 95 to 310 tons per

cubic foot (82 to 268 pounds per cubic inch or psi/in) for vertical dynamic loading of rigid machine foundations. Vesic (1965) determined by tests on large size models of steel beams (WF rigid beams 8 inches wide) resting on a subgrade of compacted micaceous silty sand (of residual origin; 37 percent passing No. 200 sieve) that k_o = 87 psi/in for static loads to about 14 psi pressure. The static shear modulus of the soil tested by Vesic was about 400 psi, and the void ratio was 1.16. The seismic modulus of the soil tested by Vesic would have been much higher, possibly by a factor of 20 to 30, which would indicate a k_o of 1740 psi/in or higher and thus in the range used for small diameter pipes in the silty sand at Catawba [1, 2, 3]

2. Static effects of displacements among structures to which the piping is attached. The Nuclear Service Water piping penetrates structures supported on continuous rock -

   a. Auxiliary Building

   b. Diesel Generator Buildings

   c. NSW/SNSW Pump Structure

   and structures supported on partially weathered rock or earth -

   d. NSW Intake Structure (lake)

   e. SNSW Intake Structure (pond)

   f. SNSW Discharge Structures.

   There is also a non-seismic connection to the Low Pressure Service Water System piping for discharge during normal operation. The appropriate differential movements of the structure during the earth quake are imposed on the piping, assuming a fixed end connection at the point of entry into the structure. The method of References 6 and 7 is used to calculate the resulting stresses. An estimate for the coefficient of subgrade reaction is obtained from Reference 8.

3. Effects of gross discontinuities. The seismic stability of the site materials in which the NSW and SNSW piping is embedded is addressed by subsurface data presented in Chapter 2. Based on this, postulated effects of faulting or settlement due to vibratory ground motion are excluded from the analysis of the piping.

   The stresses calculated using the above methods, and conservatively combined, are well within allowable limits for this piping.

4. Coefficient of subgrade reaction per unit length (k_o) for site backfill material varies with pipe size and is tabulated above.

3.7.3.12.1 Buried Seismic Category I Polyethylene Material Piping Systems

The Construction Code of record for Catawba ASME Class 3 piping is the ASMW Boiler and Pressure Vessel Code, Section III, Subsection ND, 1974 Edition including Summer 1974

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[1] Vesic, A.B., "Bending of Beams Resting on Isotropic Elastic Solid," *Proc. ASCE*, Vol 87, No. EM-2, 1961.

[2] Vesic, A.B., "Beams on Elastic Subgrade and theWinkler's Hypothesis," prepared for 6th World Conference on Soils and Foundations, Montreal, 1965.

[3] Barkan, D.D., *Dynamics of Bases and Foundations*, McGraw-Hill Book Company, New York, New York, 1962.

Addendum. This Construction Code and the ASME BPV Code, 1998 edition with 2000 addenda do not provide rules for the design, fabrication, installation, examination, and testing of piping constructed using High Density Polyethylene (HDPE) material. Reference 44 describes conditions under which HDPE pipe may be used for the construction of Section III, Division 1, Class 3, buried piping systems.

Per Reference 44, 45 and 46, the NRC concluded that the use of HDPE buried pipe in ASME Code Class 3, 12-inch nominal diameter supply and return piping lines to the 1A, 1B, 2A, and 2B diesel generator jacket water coolers will provide an acceptable level of quality and safety.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

The principal means of preventing adverse interaction between seismic and non-seismic piping is physical separation. In cases where seismic and non-seismic piping connect, the design is such that the seismic boundary is protected. Effects of the non-seismic piping are simulated or included in the math model of the seismic portion. If necessary, seismic restraints are provided for a reasonable distance beyond the seismic interface or to the first anchor in the non-seismic piping.

3.7.3.14 Seismic Analyses for Reactor Internals

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 SSE levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in Reference 4.

Fuel assembly lateral structural damping obtained experimentally is presented in Reference 9 (Figure B-4). The data indicates that no damping values less than 10 percent were obtained for fuel assembly displacements greater than 0.11 inches.

The distribution of fuel assembly amplitudes decreases as one approaches the center of the core. The average amplitude for the minimum displacement fuel assembly is well above 0.11 inches for the SSE.

Fuel assembly displacement time history for the SSE seismic input is illustrated in Reference 9 (Figure 2-3).

The CRDM's are seismically analyzed to confirm that system stresses under the combined loading conditions as described in Section 3.9.1 do not exceed allowable levels as defined by the ASME Code, Section III for upset and faulted conditions. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation and resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses are then combined with the stresses from the other loadings required and the combination is shown to meet ASME Code, Section III requirements.

<u>Duke Fuel Assembly Compatibility Evaluation for the supply of 17x17 Westinghouse Robust Fuel Assemblies</u>

<u>Seismic Evalution</u>

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 is given in Section 3.9.1.4.5.2 Reactor Vessel and Internals Modeling.

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 accerations used in Catawba RPV system analysis were based on the seismic response spectra. The time history accelerations were developed using DEBLIN2 Computer Code, reference 39. 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 40. 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 41. The spectra corresponding to the synthesized time history motions meet the acceptance criteria given in Safety Review Plan (SRP) 3.7.1 reference 42. 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 plant, 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 used for the fuel/grid impact analysis. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in Reference 4.

3.7.3.15 Analysis Procedure for Damping

For seismic piping other than NSSS, damping values used in the analyses are discussed in Section 3.7.1.3. This piping is analyzed as seismic subsystems.

Analysis procedures for damping for subsystems in Westinghouse's scope of responsibility are given in Sections 3.7.1.3 and 3.7.2.1.

3.7.4 Seismic Instrumentation Program

3.7.4.1 Comparison with NRC Regulatory Guide 1.12

Seismic instrumentation has been provided to acquire site seismic event (earthquake) response data. This instrumentation conforms with the intent of the Regulatory Guide 1.12, Revision 2 guidance, to the extent that implementation is consistent with the original licensing bases for the Catawba design. Design implementation excludes the installation of a free field sensor as justified per Section 3.7.4.2. In addition, the guidance of Regulatory Guides 1.166 and 1.167 (which are referenced per Revision 2 of Rgulatory Guide 1.12) remain outside the Catawba

current licensing bases scope, relative to pre-event planning and post-event assessments processes.

The current instrumentation system reflects the best-available state-of-the-art hardware (at time of its installation), which provides for system design specifications and functional capabilities that are comparable to the applicable Instrumentation Characteristics recommended by Regulatory Guide 1.12, Revision 2, Section C.4.

3.7.4.2 Location and Description of Instrumentation

Five strong motion accelerographs are installed within Unit 1 structures, since the same seismic response at each Unit is expected from a given earthquake. Each accelerograph is a solid-state assembly consisting of a triaxial accelerometer sensor with a digital time-history recorder and integral seismic trigger. Sensor orientation for all accelerographs is identical. Each accelerograph has the capability to measure and digitally record/store the seismic event conditions, and has the capability for local data retrieval, if necessary.

The seismic instrumentation system also consists of a network control center (NCC), which is used for rapid interrogation of the accelerograph data and for data transfer to a dedicated system computer for subsequent data processing and analysis. The time-history recorded at each accelerograph location can be analyzed to determine its corresponding peak acceleration values and to verify that site Operating Basis Earthquake (OBE) limits have not been exceeded. In addition, the recorded seismic data can also be analyzed using a resultant frequency domain event response spectrum, which can be overlayed for comparison against the Catawba design OBE and Safe Shutdown Earthquake (SSE) reference spectra.

Each of the following NCC system control and indication functions will initiate once its corresponding accelerograph "trigger" setting is sensed: high speed recording of all accelerographs starts at or above a conservatively-defined acceleration threshold (e.g., 0.01 g); and the OBE Exceedence indication via its control room alarm occurs at or above acceleration levels which define an OBE condition (e.g., 0.08g, as defined per Section 3.7.1).

The major Category I structures, Reactor Building, Containment and Auxiliary Buildings are founded on a common rock foundation (as discussed in Sections 3.7.1.4 and 3.7.2.4) and have similar base motions. The dynamic structural properties and responses of these structures are generated using similar assumptions and analytical techniques. While the response of these structures could be determined based upon the instrumentation in one structure, additional (low-exposure) Category I building locations were selected given regulatory guidance. The Catawba installed system configuration is summarized (or excepted) as follows in relation to NRC location recommendations (from Section C, Paragraph 1.2 of Regulatory Guide 1.12, Revision 2):

1. Free-field:

   Not applicable for the Catawba design and licensing bases, based upon the original site design implementation characteristics. ANSI N18.5-1974, Section 4.1.1 and subsequent ANSI/ANS-2.2-1988, Section 4.1.1 recommended that a single instrument may be located at the free field or the containment structure foundation, if soil structure interaction is negligible. Since major seismic Category I structures at Catawba are founded on rock, top of soil (free field) responses do not provide useful analytical data for the evaluation of major Category I structures founded on rock. Therefore, it is felt that free field instrumentaion does not contribute to the evaluation of these structures.

   NRC use of the free field sensor (per Regulatory Guide 1.12 Revision 2) provides Cumulative Absolute Velocity (CAV) criteria for the top-of-soil as a more realistic check

of OBE Exceedence in accordance with RG 1.166, Section 4.2. However, given that major Catawba Category I structures are founded on rock, and since CAV exceedence criteria is only applicable for structures founded on soil, the OBE Exceedence Check for Catawba will be based upon sensors mounted on the foundations of Category I structures and the applicable Response Spectrum Check from RG 1.166, Section 4.1.

2. Containment foundation:

   One accelerograph is located at the Reactor Building foundation in the annulus area outside the Containment (relative to floor elevation 555'-2"). Given its foundation mounting location, the above-noted trigger actuations are developed from this instrument.

3. Two elevations (excluding the foundation) on a structure inside the containment:

   Two accelerographs are positioned in the Upper Containment structure, to coincide with lumped mass points used in the Reactor Building structure analytical model. One is installed on the concrete steam generator enclosure wall (near operating deck floor elevation 605'-10"). Another is installed on top of the steam generator enclosure (near grating elevation 651'-0"), on the stiffener slab between the divider wall and the crane wall.

4. An independent Seismic Category I structure foundation where the response is different from that of the containment structure:

   One instrument is located at the Auxiliary Building foundation in the Auxiliary Feedwater Pump Room (relative to floor elevation 543'-0"). Given its foundation mounting location, the above-noted trigger actuations are developed from this instrument.

5. An elevation (excluding the foundation) on the independent Seismic Category I structure selected in 4 above:

   One instrument is located inside the main control board structure (relative to floor elevation 594'-0"). The above-noted trigger actuations are developed from this instrument, as a confirmatory measure that the seismic event was "felt" by the control room operators.

3.7.4.3 Control Room Operator Notification

Immediate control room alarm indication of an earthquake of 0.08 g or greater is annunicated through the system's network control center (NCC), following seismic trigger actuation by at least two accelerographs at locations 2, 4, or 5 (as described in Section 3.7.4.2 above). In addition, plant computer status indications are provided from the NCC to identify the start of accelerograph recorders and various accelerograph/system-related "trouble" conditions.

3.7.4.4 Comparison of Measured and Predicted Responses

In the event of an earthquake, the measured/recorded seismic event data is analyzed to determine the magnitude of the earthquake in terms of its peak acceleration and its frequency content. This event data is superimposed over the predicted plant design response spectrum for an OBE. Based upon the prompt determination that the OBE design response spectrum was exceeded, and coupled with plant walkdown information regarding the condition of required shutdown systems, the Units are shut down in an orderly fashion (if still operating following the seismic event). Structures, systems, and equipment will be thoroughly investigated prior to restart of the Units.

3.7.4.5 Test and Inspection

Calibration and alignment are typically performed on this instrumentation outside the fuel loading evolution in order to assure proper system operation. Periodic testing and calibration is performed in accordance with the Selected Licensee Commitments. Instrumentation functionality requirements are also addressed by the Selected Licensee Commitments.

3.7.5 References

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3.8 Design of Category I Structures

3.8.1 Concrete Containment

The concrete containment is a free standing structure separate and independent of the steel containment vessel. The concrete containment structure is referred to as the Reactor Building.

Note:

This section of the FSAR contains information on the design basis and design criteria for the Reactor Building **Concrete Containment**. Additional information that may assist the reader is provided in Design Basis Specification for the Reactor Building Structures (CNS-1144.00-00-0010).

3.8.1.1 Description of the Containment

The Reactor Building is a reinforced concrete structure composed of a right cylinder with a shallow dome roof and flat circular foundation slab. The cylinder has an inside radius of 63 ft 6 in. and a wall thickness of 3 ft. The dome has an inside spherical radius of 87 ft and is 2 ft 3 in. thick. The foundation slab is 137 ft in diameter and 6 ft thick. The structural outline of the Reactor Building is shown in Figure 3-243. Additional Reactor Building plans and sections are shown in Figure 1-10 through Figure 1-18. The structural acceptance criteria for the Reactor Building are discussed in Section 3.8.1.5.

The Reactor Building houses the steel containment vessel and is designed to provide environmental as well as missile protection for the steel shell. Section 3.8.2 completely describes the steel containment vessel and defines the criteria for its design.

A six foot annular space is provided between the steel containment vessel and the Reactor Building for control of the containment external temperatures and pressures. The annular space also provides a controlled air volume for filtering and provides access to penetrations for testing and inspection. Following a loss-of-coolant accident, the annular space is kept at a slightly negative pressure to control and filter radioactive leakage, if any, from the containment vessel and penetrations. A detailed description of the Annulus Ventilation System is found in Section 6.2.

3.8.1.1.1 Foundation Slab

The foundation slab is primarily reinforced using #18 and #11 grade 60 reinforcing steel. The reinforcing pattern consists of #18 bars placed radially and #11 bars placed circumferentially. Figure 3-244 through Figure 3-246 show the reinforcement arrangement for the foundation slab.

The containment vessel is anchored to the foundation slab. The bottom of the containment consists of a 1/4 in. thick steel liner plate. This containment liner plate is welded to embedded tees in the foundation slab. Figure 3-247 shows the liner plate connection details.

Anchorage of the containment vessel cylindrical wall to the foundation slab is accomplished by using embedded anchor bolts as shown in Figure 3-247.

The anchorage of containment interior structures is accomplished by cadwelding rebar to either side of a thickened portion of the liner plate. Figure 3-247 shows this anchoring scheme for the crane wall. Other smaller structures as well as components such as the steam generators and reactor coolant pumps are anchored in a similar fashion using bars or plates welded to each

side of the liner. Figure 3-247 shows typical anchors for the steam generators and other interior components.

3.8.1.1.2 Cylindrical Wall

The cylindrical wall is primarily reinforced using #11 rebar grade 40 or 60. The reinforcing arrangement is shown in Figure 3-248 through Figure 3-251.

Piping penetrations and openings for personnel access in the cylindrical walls are housed within surrounding Auxiliary Building structures. The equipment hatch opening has a three foot thick removable concrete cover mounted on a track, and rigidly attached to the Reactor Building during operation. Thus the penetrations do not impair the Reactor Building function of environmental and missile shielding of containment.

The only externally attached structure is the station vent stack from elevation 611+0 to elevation 718+9 at azimuth 227°-30'. The internally attached structures are miscellaneous platforms, ladders, cable trays, and pipe supports, etc.

3.8.1.1.3 Dome and Ring Girder

The principal reinforcing for the dome and ring girder is #11 and #10 rebar as shown in Figure 3-252 through Figure 3-254.

Table 3-36 provides the design moments, shears, and reinforcements provided for critical areas of the shell wall. In general, a KALNIN shell program and Q-THETA program was used to obtain a deck of cards with stress resultants for the shell. These cards were the input for a fortran program which designs required areas of steel. Figure 3-256 depicts the model used in the Reactor Building shell wall analysis. Figure 3-257 depicts the axis orientation.

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. Articles in Section III, Division 2 of the ASME Boiler an Pressure Vessel Code and Regulatory Guides dealing with concrete containments are not applicable to the Catawba Reactor Building, which is not the containment vessel. The steel containment vessel is discussed in Section 3.8.2.

Conformance to Regulatory Guides 1.15, 1.55, 1.94, and 1.142 is discussed in Section 1.7.

The concrete design at Catawba complies with ACI 318 rather than ACI 349. ACI 318 complies conservatively with ACI 349 except for the load combinations. The loads and load combinations used in the design are provided in Table 3-32. The load combinations in this table are conservative when compared to those of ACI 349, Section 9.3.

Although, based on its implementation date, Regulatory Guide 1.142 is not applicable to Catawba, a comparison of the requirements of ACI 318 and ACI 349 is provided in Table 3-35.

3.8.1.3 Loads and Load Combinations

The loads and load combinations used for the design of all Category I structures including the Reactor Building are tabulated in Table 3-32.

The specific loads that the Reactor Building is designed for are as follows:

1. Dead Loads
2. Operating Loads

   a. Live Loads
   b. Snow and Ice Loads
   c. Penetration Loads and Pipe Reactions
   d. Soil and Water Pressure
   e. Thermal Loads
3. Construction Loads
4. Wind Loads
5. Tornado Loads
6. Seismic Loads
7. Accident Loads
   a. Pipe Rupture Loads
   b. Thermal Loads
   c. Internal Negative Pressure

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 plant, which include normal thermal loads and penetration loads due to pipe reactions.

All Category 1 structures are designed for snow and ice loads, refer to Table 2-31. Those structures having roof drains are designed for the effects of frozen drains resulting from the 48 hour winter precipitation event.

Construction Loads

Construction loads include the effects of temporary loads that may occur prior to completion of the structure. These loads are evaluated, if applicable, for the partially completed structure.

Wind Loads

The design wind loads are defined in Section 3.3.1.

Tornado Loads

Tornado Loads are defined in Section 3.3.2.

Seismic Loads

Seismic loads are defined in Section 3.7.

Accident Loads

Some of the piping is attached to the Reactor Building Penetrations and therefore pipe rupture loads may be transmitted to the Reactor Building walls. The Annulus Ventilation System produces a slight negative internal pressure in the annulus. The Reactor Building is therefore analyzed for a vacuum pressure of 0.62 psi.

3.8.1.4 Design and Analysis Procedures

The Reactor Building is analyzed using Kalnin's computer program (Reference 1) for axisymmetric shells. The analytical techniques used by the program are given in Section 3.8.2.4.

The non-axisymmetric loads considered in the design of the Reactor Building are the normal and tornado wind loads, and soil and hydrostatic loads. The wind loads are analyzed by approximating the wind distribution 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 soil and hydrostatic loads are similarly represented using a Fourier Series.

The individual Fourier harmonics are analyzed by the Kalnin program and combined to produce the total stress resultant for the series.

The seismic analysis is also performed using Kalnin's program. A description of this analysis is found in Section 3.7.2.1.1.

In addition to the overall Reactor Building analysis, a further more detailed analysis is performed for the areas surrounding large penetrations, and the areas surrounding small penetrations that carry loads.

These areas are modeled using the finite element capabilities of the STRUDL computer program. An example of the models used is shown in Figure 3-255. A large enough portion of the structure is modeled so that the boundary conditions do not greatly influence the stress distribution around the opening. Reinforcing steel is then designed for the stress concentrations around these openings.

3.8.1.5 Structural Acceptance Criteria

The Reactor Building is designed in accordance with the provisions of ACI 318-71. The load combinations used in the design and the section strengths required to resist these load combinations are listed in Table 3-32.

The ultimate strength design method is used for all loading combinations for both service and faulted conditions. The required strength U will include the appropriate capacity reduction factor ϕ as defined in Section 9.2 of ACI 318-71.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

NOTE: The dates for the various ACI Specifications and ASTM Standards given in section 3.8.1.6 reflect the date of record at the time the FSAR was originated. Subsequent procurement and placement of reinforcing and concrete may be performed using more current specifications and standards (reference PIP C09-02345).

3.8.1.6.1 Materials

The Reactor Building is composed of poured in place, reinforced concrete with miscellaneous embedments fabricated from structural steel.

3.8.1.6.1.1 Concrete

The basic ingredients of concrete are cement, fine aggregate, coarse aggregate, and water. Admixtures are used as required for individual mix designs.

Cement is Type I conforming to "Specifications for Portland Cement", ASTM C150-72.

Fine aggregate conforms to "Specifications for Concrete Aggregates", ASTM C33-74a.

Coarse aggregate conforms to one of the following:

1. "General Requirements for Aggregate" and "Aggregate for Portland Cement Concrete", Sections 905 (Revised July 29, 1975) and 914-2 (Revised October 5, 1976), respectively, of the 1972 North Carolina Department of Transportation's *Standard Specifications for Roads and Structures.*
2. "Coarse Aggregate for Portland Cement Concrete for Structures", Section 701.07 of the 1973 South Carolina State Highway Department's *Standard Specifications for Highway Construction.*
3. "Specifications for Concrete Aggregates", ASTM C33-74a.

Mixing water is evaluated by performing the following tests:

1. Soundness, in accordance with "Autoclave Expansion of Portland Cement", ASTM C151-74a. The results expressed in percent obtained for the mixing water do not exceed by more than .10 those obtained for distilled water.
2. Time of setting, in accordance with "Time of Setting Hydraulic Cement by Vicat Needle", ASTM C191-74. The results obtained for the mixing water are not less than 45 minutes for initial setting time and not more than 8 hours for final setting time.
3. Compressive strength, in accordance with "Compressive Strength of Hydraulic Cement Mortars (Using 2 in. Cube Specimens)", ASTM C109-73. The 7 and 28 day strengths of mortar cubes made with the mixing water are not lower by more than 10 percent of the strengths obtained for mortar cubes made with distilled water.

The water used to make ice conforms to the requirements for mixing water described above.

Admixtures, when used as determined by individual mix design, are in conformance with the applicable ASTM Standard as follows:

1. Air-entraining admixtures conform to "Specifications for Air-Entraining Admixtures for Concrete", ASTM C260-73.
2. Water reducing, retarding, and accelerating admixtures conform to "Specifications for Chemical Admixtures for Concrete", ASTM C494-71.
3. Pozzolanic admixtures conform to "Specifications for Fly Ash and Raw or Calcined Natural Pozzolans for use in Portland Cement Concrete", ASTM C618-77, except that the maximum value for loss on ignition is to be 6.0 percent.
4. Slag cement conforms to "Specifications for Blended Hydraulic Cements", ASTM C595-74.
5. Superplasticizing admixtures shall conform to "Specifications for Chemical Admixtures for Concrete," ASTM C494-77 for a type A admixture.

The combined chloride content of the admixtures and mixing water does not exceed 250 ppm.

Ingredient materials are stored in accordance with the detailed recommendations presented in Chapter 2 of "Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete", ACI 304-73.

Concrete mixes are designed in accordance with Chapter 3 of "Specifications for Structural Concrete for Buildings", ACI 301-72 with ingredients proportioned by method 1 of Section 3.8.2.1 of ACI 301-72. Concrete for floors conforms with the mixes designated on the design

drawings and the provisions of Section 3.8.5, ACI 301-72 do not apply. The slump of concrete for floors meets the requirements of Section 3.5 of ACI 301-72.

The various concrete mixes that are used in the Reactor Building and throughout the plant are listed in Table 3-33.

The batching, mixing, and transporting of concrete conforms to Chapter 7 of "Specifications for Structural Concrete for Buildings", ACI 301-72 except for the following:

1. The frequency of calibration of scales used for weighing concrete ingredients is specified by the Quality Assurance Department.
2. Water may be added to the concrete at the point of placement when approved by the Project Engineer - Construction, but the total water may not exceed 101 percent of the design mix. When additional concrete is added, additional mixing is performed in accordance with "Specifications for Ready-Mixed Concrete", ASTM C94-74.

The placement of concrete consisting of preparation before placing, conveying, depositing, protection and bonding is in accordance with Chapter 8 of "Specifications for Structural Concrete for Buildings", ACI 301-72. The following exceptions are taken to this section:

1. During hot weather, the temperature of the plastic concrete, as placed, does not exceed 85°F.
2. Adjacent concrete pours in contact have a minimum interval of time between pours of three days unless otherwise specified or approved by the Design Engineering Department.
3. Where bond is required in construction joints, the method and requirements for providing bond are specified on drawings.

3.8.1.6.1.2 Reinforcing Steel

Reinforcing steel is deformed reinforcing bars conforming to "Specification for Deformed and Plain Billet-Steel Bars for Concrete Reinforcement", ASTM A615-72, Grades 40 and 60. The fabrication of reinforcing bars is in accordance with Concrete Reinforcing Steel Institute (CRSI) "Manual of Standard Practice", dated January 1, 1973, Chapter 7. Placement of reinforcing steel conforms to the general requirements of Chapter 5 of "Specifications for Structural Concrete Buildings", ACI 301-72.

Reinforcing steel is spliced using either lap splices meeting the requirements of ACI 318-71 or Cadweld meeting the requirements of Regulatory Guide 1.10.

3.8.1.6.1.3 Liner Plate and Associated Anchors and Hardware

The materials used for the liner plate, its anchors and associated hardware are listed in Table 3-40. A complete discussion and description of the containment portion of structure is given in Section 3.8.2.

3.8.1.6.1.4 Structural Steel

Structural steel used for embedments consists of low carbon steel shapes, plates, and bars conforming to "Standard Specifications for Structural Steel", ASTM A36, ASME SA-306 Gr.60, ASTM A-588, or the materials listed in Table 3-40.

3.8.1.6.2 Quality Control

Concrete quality is controlled by sampling and testing both the ingredients and the mixed concrete as placed.

Cement is tested for conformance with ASTM C150-72. A sample is taken a minimum of once every 1100 tons.

Aggregate sampling is performed in accordance with "Sampling Aggregates", ASTM D75-71. The tests to be performed and the interval of testing is shown in Table 3-34. Test results for fine aggregate are evaluated in accordance with ASTM C33-74a. Test results for coarse aggregate are evaluated in accordance with one of the references listed in Section 3.8.1.6.1 for coarse aggregate. An additional test is performed on coarse aggregate annually to determine flat and elongated particle content. This test is performed in accordance with "Method of Test for Flat and Elongated Particles in Coarse Aggregate", CRD-C119-53. No more than 15 percent by weight is allowed for flat and elongated particles.

Mixing water is sampled and tested for the requirements of Section 3.8.1.6.1 at intervals of six months.

Fly ash and pozzolans are sampled and tested in accordance with ASTM C311-77, and these materials must meet the requirements of ASTM C618-77.

Admixtures are tested in accordance with ASTM C494-71 and ASTM C260-73 by the suppliers. Test reports are submitted with each shipment and reviewed by the Quality Assurance Department.

Central-mixed and truck-mixed concrete is sampled and tested for uniformity at six month intervals in accordance with ASTM C94-74. Also, unit weight and yield of freshly mixed concrete is determined daily during production.

During construction, concrete is sampled in accordance with ASTM C172-71 for various quality control tests. For every 100 cubic yards or fraction thereof placed in one day for each mix, six compressive strength test specimens are made in accordance with ASTM C31-69. At the same sampling interval, air content tests are performed in accordance with ASTM C231-73. For every 50 cubic yards or fraction thereof placed on one day for each mix, slump tests are performed in accordance with ASTM C143-71, and the concrete temperature in measured.

The quality of other materials such as reinforcing steel and structural steel is controlled by requiring suppliers to furnish appropriate mill test reports which are reviewed and approved in accordance with the general provisions of the overall Quality Assurance Program outlined in Chapter 17.

3.8.1.6.3 Special Construction Techniques

No unique or untried construction techniques are used. The same construction methods are used that have been successfully used at McGuire Nuclear Station which is identical in shape, size, and form.

3.8.1.7 Testing and Inservice Inspection Requirements

A visual inspection of the exposed accessible interior and exterior surfaces of the Reactor Building is performed three times every ten years coinciding with containment visual examinations required by Technical Specifications Surveillance Requirement 3.6.1.1.

3.8.2 Steel Containment

Note:

This section of the FSAR contains information on the design basis and design criteria for the Reactor Building **Steel Containment**. Additional information that may assist the reader is provided in Design Basis Specification for the Reactor Building Structures (CNS-1144.00-00-0010).

3.8.2.1 Description of the Containment

The containment vessel is a freestanding welded steel structure consisting of a vertical cylinder with a hemispherical dome and a flat circular base. The cylinder is stiffened by circumferential ring girders and vertical stringers welded to the exterior of the vessel. The containment shell is anchored to the Reactor Building foundation by means of anchor bolts around the perimeter of the cylinder base. The flat base of the containment is a 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 diameter of 115 ft and overall height of 171 ft 3 in. Further details of the containment geometry and Reactor Building general arrangements are shown in Figure 3-243 and Figure 1-10 through Figure 1-18. Additional containment vessel details, including plate thicknesses and vertical and circumferential stiffener locations, are shown in Figure 3-259, Figure 3-262 and Figure 3-263. Containment vessel anchorage details are shown in Figure 3-247.

Several penetrations are required through the containment vessel for personnel and equipment access, fuel transfer and various piping systems. The containment penetrations are:

1. EQUIPMENT HATCH

   The equipment hatch is composed of a 20 foot cylindrical sleeve in the containment shell and a dished head with mating, bolted flanges. The flanged joint has double compressible seals with an annular space for pressurization and testing.

   The equipment hatch is designed, fabricated and tested in accordance with Section III, Subsection NE, of the ASME Boiler and Pressure Vessel Code.

   Details of the equipment hatch are shown on Figure 3-260.

2. PERSONNEL LOCKS

   Two personnel locks are provided for each unit. Each lock consists of a cylindrical sleeve with a bulkhead at each end. Each bulkhead has double gasketed doors with an interlocking system to prevent both doors from being opened simultaneously. Instrumentation is provided to indicate the position of each door. Double, inflatable seals are provided on each door. Automatic leak testing equipment is piped to the annular space between seals for periodic testing as required by the Technical Specifications. The use of double inflatable seals allows testing of the annular space without the use of external strong-backs or other remote devices.

   The personnel locks are designed, fabricated, tested and stamped in accordance with Section III, Subsection NE of the 1971 ASME Code, including all addenda up to the Summer of 1972. The personnel locks are installed as a unit with accompanying reinforcing collars. Details of the personnel locks are shown on Figure 3-260.

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 3-261 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 a welded pipe cap.

5. PENETRATION SLEEVES

All penetration sleeves are preassembled and welded into containment vessel shell plates as shown on Figure 3-262 and Figure 3-263. Each shell plate having penetration sleeves is stress relieved prior to installation into the containment.

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 3-261.

7. ELECTRICAL PENETRATIONS

Medium voltage electrical penetrations for reactor coolant pump power (shown on Figure 3-261) 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 8.1.

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 penetration types may be used for this verification if the particular environmental conditions of the test are equal to or exceed those for Catawba.

8. MECHANICAL PENETRATIONS

Typical mechanical penetrations are shown on Figure 3-261.

Mechanical penetration functional requirements, code considerations, analysis and design criteria are defined in Section 3.6.2.4.

A comprehensive list of all penetrations is given in Figure 3-264.

3.8.2.2 Applicable Codes, Standards and Specifications

The containment vessel is designed, fabricated, constructed and tested in accordance with Subsection NE, Section III, of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including all addenda through the Summer of 1972.

Subsection NE, 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 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.1.b - Add liquid penetrant method as an acceptable means of testing liner seam welds.

   Liquid penetrant is used more successfully in detecting circular defects.

2. Delete C.1.c.

   Non-destructive testing as required in C.1.b and leak chase pressure testing to peak containment pressure as required in C.1.d have been successful in detecting leaks in seam 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.1.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 to detect leaks created during construction activities after the completion of the initial leak chase pressure test and during placing of concrete around the chase system.

3. In C.1.d where leak chase system channels are installed over liner welds, the system is tested for leak tightness by pressurizing to minimum of 12.5 psi pressure. If any leak is indicated in a 30-minute period, the leak is repaired and the system retested.

The material, fabrication (except for weld details) and allowable stresses for the transition torus meet the requirements of Subsection NE, Section III of the ASME Code, 1971 Edition including all addenda through the Summer of 1972.

The welds and test channels on the torus are as shown in Figure 3-265. The torus is considered part of the bottom liner plate and all welds are tested as specified in Section 3.8.2.7.

The base liner plates are ultrasonically tested in accordance with Subparagraph NB-2532.1 of Section III of the ASME Code to insure that there are no laminations that could result in a failure to transmit loads normal to the liner plate such as would occur in the region of B-series Cadweld Splices.

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. Thermal loads.
3. Seismic loads.
4. External pressure.
5. Design basis accident.

6. Localized loads.

Dead Load and Construction Loads

The dead load includes the weight of the containment shell and all permanent attachments. Construction loads include all loads imposed on the containment shell during construction.

Thermal Loads

The containment shell is subject to thermal loads during normal operation of the unit. The operating temperatures are between 75°F and 100°F in the upper containment compartment and between 100°F and 120°F in the lower containment compartment.

Seismic Loads

The seismic loads are derived from horizontal and vertical ground response spectra. The horizontal ground motion is represented by OBE (.08g) and SSE (.15g) spectra. The spectrum used for the vertical direction is equal to two-thirds of the horizontal spectrum. The ground motions are taken to act in one of two perpendicular horizontal directions simultaneously with the vertical direction. For further details on seismic loads see Section 3.7.

External Pressure

The external pressure load is due to 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.

Design Basis Accident

Loads for the Design Basis Accident are the result of a rupture in the primary coolant system up to and including a double-ended rupture of the largest pipe or a main steam line break. The following loads are associated with the DBA:

1. The design internal pressure is 15 psig. This pressure is applied as a static load case.
2. The containment shell must also be designed for the short-term pressure transient immediately following a LOCA or MSL break. This is a dynamic load of an asymmetric nature (i.e., the pressure varies around the circumference of the shell) due to the geometric layout of an ice condenser containment.
3. The containment atmosphere temperatures after a LOCA or MSL break are described in Section 6.2.1.
4. The containment shell experiences hydrostatic loads at the base due to post-LOCA flooding.

See Chapter 6 for further details of the Containment Design Basis Accident.

Localized Loads

Penetration loads, piping loads and jet impingement loads are all localized loads applied to the containment vessel. Penetration and piping loads are due to dead load or pipe reactions at penetrations and pipe supports welded to the shell. Jet impingement loads are due to high pressure fluid jets caused by the rupture of small diameter piping adjacent to the containment vessel.

Load Combinations

The loads applied to the containment vessel are combined in accordance with procedures given in the Standard Review Plan (see Reference 5). These combinations are summarized in Table 3-37. The nomenclature used in this table is defined in Table 3-32. Compliance with Regulatory Guide 1.57 is discussed in Section 1.7.

3.8.2.4 Design and Analysis 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.

3.8.2.4.1 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.

The containment vessel utilizes the ice condenser concept for energy absorption during a loss-of-coolant accident. This rapid energy absorption capability maintains the containment vessel design pressure at a low level as well as reducing the peak duration. See Section 6.7 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 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 Reactor Building.

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 waterproofing is provided.

The containment bottom liner plate is designed to function as a leak-tight membrane and is not required to function as a structural component. The liner plate is 1/4 inch thick carbon steel with the total thickness available for corrosion allowance. Liner plate details are shown on Figure 3-265.

3.8.2.4.2 Design Per ASME Code

The containment vessel is designed to satisfy the requirements of the ASME Code, paragraphs NE-3133 and NE-3324. These sections of the code describe the procedures to be followed in sizing the various parts of the containment shell under external pressure loading and internal pressure loading, respectively.

The localized areas around large and small openings of the containment vessel are analyzed and designed to meet the requirements of paragraph NE-3330 of the ASME Code. A systematic numerical procedure is set up in order to analyze small penetration reinforcement in accordance with the ASME Code requirements. The numerical procedure provides detailed requirements for reinforcing the shell around an opening. Separate analysis and design are performed on the equipment hatch and personnel air lock penetrations.

3.8.2.4.3 Static and Seismic Load Analysis

The containment vessel is analyzed to determine all membrane forces, moments and shears as a result of all specified static and seismic loadings. The vessel is idealized as a thin shell of revolution. The stresses and deflections produced in the shell under the applied loads are

calculated by a computer program written by Professor A. Kalnins of Lehigh University (Reference 1). The computer mathematical model used to represent the containment shell is shown in Figure 3-267. See Section 3.7.2 for further details of the seismic analysis.

3.8.2.4.4 Analysis of Local Areas

The overall analysis of the containment vessel as described in Section 3.8.2.4.3 does not examine localized stresses around the equipment hatch and personnel locks since these large penetrations cannot be accurately represented in an axisymmetric model. A more detailed analysis of this region of the shell is required. A three-dimensional finite element model of the containment was analyzed using the STARDYNE computer code to obtain results around the equipment hatch and airlocks (see Reference 3). The finite element model used for this analysis is shown in Figure 3-268.

Concentrated forces and moments are applied locally to the containment vessel by welded attachments, penetrations and jet impingement. These loads are analyzed as equivalent static loads using finite element models with the STRUDL computer code (see Reference 4). The stresses resulting from the localized loads are combined with the general membrane stresses to determine the total stress state in the shell.

3.8.2.4.5 Analysis of Design Basis Accident

The loadings on the containment vessel due to the design Basis Accident are described in Section 3.8.2.3. All of the loads except the short term pressure transient are considered as static loads and are analyzed as described in Section 3.8.2.4.3. The stress resultants and displacements of the containment shell due to an asymmetric 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, due to the use of the ice condenser concept, maintains the containment vessel design pressure at a low level as well as reducing the peak pressure duration. This reduction in peak pressure results in a 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 53 compartments (see Figure 6-25 - Figure 6-30 for compartment 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 six lower compartment elements.

3. Analytical Representation of the Shell:

   The containment vessel shell is idealized with the Wilson-Ghosh computer program (Reference 2) as an assemblage of conical frusta joined together at their nodal circles. As part of the model the stiffening rings 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 has different magnitudes in the

longitudinal and circumferential directions. The modulus of elasticity used in the longitudinal direction is an equivalent value given by E_{eq}, and is

$$E_{eq} = Eh_{eq} / h_{shell}$$

Where, h_{eq} is 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. The finite element model is shown in Figure 3-269.

4. Analysis Procedures:

   To analyze the discrete containment shell, Hamilton's variational principle is used to derive the structure's equations of motion which are then 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 their 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.

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 6-25 through Figure 6-30, the compartments to be considered for the transient load analysis of the containment vessel are only those compartments in contact with the containment inner shell surface. The pressure in each of these compartments varies with time. Figure 3-270 through Figure 3-273 show typical pressure transients due to various breaks in the Reactor Coolant System, for a representative ice condenser compartment adjacent to 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 Fourier components. Both symmetrical and asymmetrical terms are used in the final Fourier representation of the pressure transient. Thirteen Fourier components (6 sin, 6 cos, 1 constant) were employed in the representation since convergence was found satisfactory. A typical comparison of the actual pressure distribution for a given time step is shown in Figure 3-274.

3.8.2.4.6 Buckling Analysis

The stability analysis of the containment vessel is performed by the methods defined in Code Case N-284 of the ASME Code (see Reference 6). The buckling capacity of the shell is based on linear bifurcation (classical) analysis. Capacity reduction factors are applied to account for the effects of imperfections and nonlinearity in geometry and boundary conditions. Plasticity reduction factors are applied to account for nonlinearity in material properties if elastic limits are reached. Overall stability, panel buckling and stiffener buckling are evaluated using bifurcation analysis of an axisymmetric shell of revolution with the BOSOR 4 computer code (see Reference 7). Local buckling capacity of the region surrounding the equipment hatch and personnel lock is based on a bifurcation analysis of a three-dimensional finite element model with the NASTRAN computer code (see Reference 8). The BOSOR and NASTRAN computer mathematical models used for the bucking analysis are shown in Figure 3-275 and Figure 3-276, respectively. The containment vessel buckling capacity is evaluated for the load

combinations defined in Table 3-37 as well as the transient dynamic pressure load due to LOCA.

3.8.2.4.7 Ultimate Capacity Analysis

The ultimate capacity of the containment vessel due to overpressurization is evaluated by a large displacement elastic-plastic analysis. The containment vessel is represented by an axisymmetric shell finite element model with the MARC computer code (see Reference 9). Actual material properties based on a statistical analysis of mill test reports are used in determining the stress-strain curve for input to the MARC program. The material properties obtained from the statistical analysis are listed in Table 3-44. Figure 3-277 shows the finite element model used for this analysis. The criteria used in determining the collapse load is given in the 1980 ASME Code, Section III, Appendix II, Article 1430. The ultimate capacity of localized areas such as the equipment hatch, personnel lock and other penetrations is also evaluated.

3.8.2.4.8 Computer Program Description

1. Kalnin's shell program (Reference 1) uses the finite difference method to solve the differential equations for a thin shell of revolution derived by E. Reissner (Reference 10). The equations are based on the linear theory of elasticity and consider both membrane and bending action in the shell. This method of analysis (Reference 11) has been widely used in the analysis of thin axisymmetric shells.

2. The Wilson-Ghosh computer program (Reference 2) as modified and verified at Duke Power Company uses the finite element method to determine the stress resultants and displacements of axisymmetric structures. The applied loadings may be symmetric or arbitrary (in which case the load must be represented by a Fourier Series).

3. STARDYNE (Reference 3) is a general purpose finite element program for the analysis of linear elastic structures. It can perform both static and dynamic analyses; it is particularly useful in the solution of large dynamic problems.

4. STRUDL (Reference 4) is a general purpose finite element program for the analysis of linear elastic structures. It has the capability for both static and dynamic analysis.

5. MARC (Reference 9) is a general purpose finite element program. It is capable of linear and nonlinear analysis. A wide range of both geometric and material nonlinearities are available in the program such as large displacements, large strains, strain hardening and plasticity. The tangent modulus solution method is applied to nonlinear problems.

6. BOSOR 4 (Reference 7) is used in the analysis of shells of revolution. It can perform linear elastic analysis or nonlinear (large displacement) elastic analysis including the effects of initial imperfections in the shell geometry. Classical bifurcation analysis is used to determine the buckling load of the shell. The finite difference method is used to solve the differential equations of equilibrium.

7. NASTRAN (Reference 8) performs linear static and dynamic analysis of general structures using the finite element method. Buckling loads of a structure are determined by a classical bifurcation analysis.

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 Allowable Stress Limits

Allowable stresses have been established for each of the load combinations listed in Table 3-37. Compliance with Regulatory Guide 1.57 is discussed in Section 1.7. These limits are in compliance with Subsection NE, Section III of the ASME Code and are summarized in Table 3-38. Primary, secondary and peak stresses are considered in this table. The actual stress intensities obtained from the analysis of the containment vessel are given in Table 3-42 at various locations on the shell for comparison with the numerical values of the allowable stresses given in Table 3-41 (which corresponds to Table 3-38).

3.8.2.5.2 Buckling Safety Factors

The containment vessel cylinder and dome were evaluated separately to determine buckling factors of safety. Different load combinations (from Table 3-37) are critical in the separate parts of the shell. The factors of safety for overall stability and local buckling for the critical load combinations are shown in Table 3-43. The results presented in this table show that the containment vessel has a buckling factor of safety greater than 3.0 for normal loads and a minimum factor of safety of 2.0 for all cases involving the Design Basis Accident.

3.8.2.5.3 Containment Ultimate Capacity

The containment vessel has been analyzed to determine the maximum internal pressure to which it can be subjected without failure. Based upon linear elastic analysis, the requirements of the ASME Code, Division 1, paragraph NE-3220, Service Level C Limit, are met for a combination of dead load and an internal pressure of 45 psig. Based upon nonlinear analysis, the lower bound for the containment ultimate capacity is calculated to be 72 psig. Failure occurs due to the propagation of plasticity from the shell panels into the circumferential stiffeners. Localized areas have been analyzed to insure that they do not control the containment capacity. A summary of the results of these analyses is presented in Table 3-45.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The materials used for construction of the containment vessel are listed in Table 3-40. These materials meet the requirements of Article NE-2000 of Section III of the ASME Code. Tests to determine required material and physical properties are in accordance with NE-2000 and the appropriate material specifications.

Fabrication and erection of the containment vessel are in accordance with Article NE-4000 of Section III of the ASME Code. This includes welding procedures, procedure and operator performance qualifications, post weld heat treating and tolerances.

Nondestructive examination of welds and materials is in accordance with Article NE-5000 of Section III of the ASME Code.

The interior steel surface of the 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 integrated radiation dose is 2×10^7 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.

No unique or untried construction techniques are used in fabricating the containment vessel. The same construction procedures are used that have been successfully employed at McGuire Nuclear Station which is identical in shape and size.

3.8.2.7 Testing and Inservice Inspection Requirements

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 NE-6000. Penetrations are pressure tested as required for class NC in accordance with Section III of the 1974 ASME Code including addenda through the Summer of 1974.

2. Leakage Rate Tests

   Bottom Liner Plate: The bottom liner plate welds are inspected, prior to placing fill concrete, in accordance with the following:

   a. Dye penetrant examinations are performed in accordance with Section V of the ASME Boiler and Pressure Vessel Code.

   b. Upon completion of the dye penetrant test, the weld seams are covered with test channels and pressure tested. All detected leaks are repaired and retested. The leak test channel layout is shown in Figure 3-266.

   Personnel Air Locks and Equipment Hatch: The personnel air locks are pressurized and a Type B leak rate test is performed as described in Section 6.2.6.

   The double o-ring compression seals in the equipment hatch are tested for leakage as specified in Section 6.2.1.6.

   Containment Leakage Rate Test: Upon completion of erection including all penetrations, personnel air locks, equipment hatch, bottom liner plate and structural testing, a leakage rate test is performed on the containment as described in Section 6.2.6.

3.8.2.7.2 Operational Surveillance

3.8.2.7.2.1 Structural Integrity

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.

3.8.2.7.2.2 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 and integrity as described in Section 6.2.6 in accordance with plant Technical Specifications.

3.8.3 Concrete and Structural Steel Internal Structures of the Steel Containment

Note:

This section of the FSAR contains information on the design basis and design criteria for the Reactor Building **Internal Structures**. Additional information that may assist the reader is provided in Design Basis Specification for the Reactor Building Structures (CNS-1144.00-00-0010).

3.8.3.1 Description of the 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. These structures also provide support and restraint for all major equipment, components, and system located within the Reactor Building. The internal structures are supported on the Reactor Building foundation as shown in Figure 3-243. The internal structures consist primarily of poured-in-place reinforced concrete. See general arrangement Figure 1-10 through Figure 1-18 for outline drawings and dimensions. Further details of the internal structures follow.

3.8.3.1.1 Base Slab

The base slab (elevation 552+0) is a two-foot thick circular slab of reinforced concrete resting on the foundation mat. The Containment floor plate is sandwiched between the base slab and the foundation mat. Other building structures including the crane wall, reactor vessel cavity wall, and the cross-over leg restraints are interconnected with the base slab.

3.8.3.1.2 Reactor Vessel Cavity Wall

The reactor vessel cavity wall is an eight foot six inch thick cylindrical wall surrounding the reactor vessel and providing primary shielding. This wall carries the reactions from the reactor vessel support pads and the steam generator lower lateral support embedments as described in Section 5.4.14.

3.8.3.1.3 Upper Reactor Cavity

The upper reactor cavity wall is a four-foot thick reinforced concrete wall lined on the inside face with 3/16 inch stainless steel plate forming a part of the refueling canal.

Openings are located near the top of the cavity just below the operating floor to vent the cavity to the lower containment compartment. These vent openings allow pressure relief from a loss-of-coolant accident (LOCA) in the primary cavity. The control rod drive mechanism (CRDM) missile shield covers the cavity and is anchored with rods embedded in the reactor cavity walls.

3.8.3.1.4 Refueling Canal

The refueling canal consists of reinforced concrete floor and walls. The floor is three feet thick and the walls are three to four feet thick. The inside face of the walls and the floor are lined with a 3/16 inch stainless steel plate.

The canal joins with the upper reactor cavity, but the two are divided during operation by a three foot thick concrete gate. The canal is open at the top and forms a part of the upper compartment volume while the reactor cavity is enclosed by the gate and CRDM missile shields, and vents only to the lower compartment. During the refueling process, the gate is removed and both the canal and reactor cavity are flooded with water.

3.8.3.1.5 Crane Wall

The crane wall provides secondary shielding for the primary coolant system and supports much of the interior structural slabs and walls. Much of the major equipment including the polar crane, ice condenser, steam generators, pressurizer, and reactor coolant pumps receive all or part of their support from the crane wall. The crane wall is a three foot thick reinforced concrete cylinder with an inside diameter of 83 feet and overall height of 117 feet. Anchorage to the Reactor Building foundation is achieved by cadwelding reinforcing to either side of a thickened portion of the containment base liner plate as shown in Figure 3-247. The wall has several openings as required by the ice condenser, piping and cable penetrations, ventilation, and access. Embedments are located in the crane wall for steam generator, pressurizer, and reactor coolant pump supports as well as pipe and cable tray supports and miscellaneous platforms and ladders.

3.8.3.1.6 Steam Generator Compartments

The steam generator compartments are enclosures around the steam generators forming an extension of the lower containment compartment projecting above the operating floor. The steam generator enclosures are paired-up with a common divider wall of three-foot thick reinforced concrete and outside walls of two-foot thick reinforced concrete on each side. The divider wall has two openings eight feet by fourteen feet so that a steam line break in one compartment will vent to the adjacent compartment. The front of each enclosure is a removable cylindrical steel shell segment with a thickness of 3/4 inch. The top of each enclosure is a removable steel dome also having a thickness of 3/4 inch. The steel to concrete interface of the removable portions are sealed with gaskets to limit ice condenser bypass leakage from the lower to upper compartments during a LOCA or main steam line rupture.

3.8.3.1.7 Pressurizer Compartment

The pressurizer compartment encloses the pressurizer above the operating floor and forms and extension of the lower containment compartment. The enclosure walls and the top slab are two-foot thick reinforced concrete. The pressurizer upper lateral supports are anchored to embedments in the enclosure walls. Pressure hatches in the top slab provides access for maintenance and inspection.

3.8.3.1.8 Operating Floor

The operating floor (elevation 605+10) is the main divider barrier between the lower and upper containment compartments. It consists of a two-foot six inch thick reinforced concrete slab. The floor is supported around the outside perimeter by the crane wall, steam generator enclosures, and pressurizer enclosure. Interior support is provided by the refueling canal walls and the upper reactor cavity walls. Several pressure hatches are provided for access and pump removal. Gaskets are used to limit ice condenser bypass leakage through the hatches during LOCA. Steam generator upper lateral supports are anchored to embedments within the thickness of the operating floor.

3.8.3.1.9 Accumulator Floor

The accumulator floor (elevation 565+3) provides support for the accumulator tanks, lower containment ventilation units, and other smaller miscellaneous equipment. This floor consists of a two-foot thick concrete ring slab between the crane wall and containment. The slab is cast integrally with the crane wall and is supported around the outside perimeter by a series of structural steel columns. Although it spans to the containment, there is no interconnection with

the containment. A compressible material is placed between the slab and containment to ensure that there is no interaction.[1] See below

3.8.3.1.10 Ice Condenser Floor

The ice condenser floor (elevation 593+8 1/2) is a two foot six inch thick reinforced concrete slab similar to the accumulator floor. This slab is cast integrally with the crane wall and forms a ring between the crane wall and containment. The outside perimeter is supported by a series of structural steel columns. The structural slab is topped with a layer of foam concrete for insulation and a four inch thick wear slab on top of that.

The floor slab model was analyzed using STRUDL. The model consists of radial beams every 3°-6' with circumferential beams every 2.558'. Advantage was taken of symmetry and only a portion of the total slab was modeled. Proper considerations were made for the boundary conditions of this partial model representation.

The ice condenser support columns were included in the model as elastic supports and were analyzed as pinned end columns. The column are bolted to the floor with A36 bolts placed in embedded sleeves through the floor. (For a more detailed description of the ice condenser cavity, refer to Section 6.7).

Load combinations were made per Table 3-32 per SRP 3.8.4. The floor slab was designed per ACI 318-71 and the support column were designed per ASCI, 7th edition.

3.8.3.1.11 Control Rod Drive Mechanism (CRDM) Missile Shield

The CRDM missile shield is a series of interlocking removable concrete beams located over the reactor cavity at the operating floor. The beams are three feet thick and are plated on the bottom side with one inch thick steel in the area of postulated control rod ejection. The beams are anchored to the floor by means of rods embedded into the walls of the reactor cavity.

3.8.3.1.12 Refueling Canal Gate

The refueling canal gate consists of three foot thick removable concrete wall sections dividing the reactor cavity from the refueling canal. The gate sections span laterally between seven and a half-inch deep slots formed vertically in the refueling canal walls.

3.8.3.1.13 DELETED

3.8.3.1.14 NSSS Support Systems

The support systems for the reactor vessel, steam generators, reactor coolant pumps, and main loop piping are completely described in Section 5.4.14.

3.8.3.1.15 Accumulator Wing Walls

The accumulator wing walls are two foot thick radial walls on either side of the accumulator tanks. They are doweled to the crane wall, accumulator floor, and ice condenser floor forming an enclosure around each accumulator tank.

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[1] The compressible material may be removed permanently to allow for the Steel Containment Vessel (SCV) coatings inspections and/or repair work.

3.8.3.1.16 Ice Condenser End Walls

The ice condenser end walls are two foot thick radial walls enclosing the ice region outside the crane wall. They extend from the ice condenser floor to the top of the crane wall. Dowels are provided to both the ice condenser floor and the crane wall. Pressure doors for equipment access are located near the top of each of the two walls. A smaller personnel access pressure door is provided at the bottom of one of the walls.

3.8.3.1.17 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 design drawings (Ref. Figures 3-278 and 3-279) as follows:

Ice Condenser Seals	Seal MK S10 and Seal MK S11
Steam Generator Enclosures Seals	Seal MK S7, Seal MK S8 and Seal MK S9
Operating Deck Hatches and Access	Seal MK S1 through MK S6
Opening Seals	Seal MK S18 and Seal MK S23
Refueling Canal Seals	Seal MK S17
Pressurizer Enclosure Seals	Seal MK S26 through MK S29

Note: For more detailed information refer to Figures 3-278 and 3-279.

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 Figure 3-278 through Figure 3-279. 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 3-46 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.

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.

3.8.3.2 Applicable Codes, Standards, and Specifications

The interior structures are designed according to the applicable codes and specifications listed in Table 3-31. The supports and restraints for the NSSS components are designed by the criteria set forth in Section 5.4.14.

The materials, quality control, and construction techniques are the same as those described in Section 3.8.1.6.

3.8.3.3 Loads and Load Combinations

The loads and load combinations considered for the design of the interior structures are listed in Table 3-32. These load combinations are in agreement with Standard Review Plan 3.8.3. of November, 1975. A description of the major loads follows.

3.8.3.3.1 Normal Loads

The normal loads include dead weight of the structure and equipment as well as live loads and reactions associated with piping and equipment in normal operation.

Structural components exposed to thermal gradients are designed for the maximum gradient occurring during normal operation.

3.8.3.3.2 Seismic Loads

The interior structure is designed to withstand two earthquakes. The operating basis earthquake (OBE) is to be sustained without interruption of normal operation. The safe shutdown earthquake (SSE) is to be sustained without interfering with the safe shutdown of the plant.

The seismic analysis is fully described in Section 3.7. Consideration is given to both structural and equipment seismic forces.

3.8.3.3.3 Accident Loads

The interior structure is designed for the effects of postulated pipe breaks as defined in Section 3.6. These effects include differential compartment pressure, pipe restraint reactions, and pipe rupture reactions.

Peak differential compartment pressures are determined from the compartment pressure-time histories furnished by Westinghouse for the various postulated pipe breaks. The pressure-time histories are obtained from a transient mass distribution (TMD) model described fully in Section 6.2.1.2. The TMD model consists of 53 compartments interconnected by flowpaths representing physical openings in the structure. The Westinghouse pressure-time histories are sorted by a Fortran computer program to select the maximum differential pressure occurring across the common boundary of each pair of adjacent compartments. These peak positive and negative differential pressures between compartments are listed in Table 3-47. Also listed in the table are the time after the start of the blowdown at which the peak occurs, and identification of the break producing the peak pressure. The pressures furnished by Westinghouse are increased by 40 percent for design purposes. These increased design pressures are also listed in Table 3-47.

In addition to designing the individual structural components for pressure, 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 3-48 lists the maximum values of uplift, shear and overturning moment, the time at which they occur and the break identification for which they occur.

The loadings described above were utilized in the design of the interior structure. Subsequent to this design a revised postulated pipe break criteria was introduced in Section 3.6. The differential pressures and load resultants presented in Table 3-47 and Table 3-48 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.

Many of the postulated pipe break locations are provided with restraints to limit movement and consequential damage as a result of the pipe break. The structure is therefore designed for the reactions including dynamic effects associated with the pipe restraints.

The interior structure is also designed for the jet impingement forces created when a pipe ruptures near the structure. The dynamic effect of the suddenly applied jet impingement force is also considered.

Internally generated missiles are discussed in Section 3.5.1.2. The interior structure is designed to withstand the impact of such internal missiles and the dynamic effects associated with them.

3.8.3.3.4 Other Design Criteria

The NSSS supports are designed for the load combinations and criteria set forth in Section 5.4.14. The steel portion of the divider barrier between the upper and lower compartments (consisting of the steam generator enclosures) are designed in accordance with Section III, Subsection NE, of the 1974 ASME Code including addenda through the Summer of 1976. A further discussion of the steam generator enclosures is included in Section 3.8.3.4.

3.8.3.4 Design and Analysis Procedures

The elements of the interior structure are designed on an individual basis. The interconnection between elements is included by considering relative stiffnesses of connected elements to determine boundary conditions. In some cases, portions of adjacent structural elements are modeled along with the particular element being designed to obtain the proper boundary interaction. For other cases a most conservative approach of designing for both fixed and pinned boundary conditions is used. A complete description of structural models follows.

3.8.3.4.1 Base Slab

The base slab at elevation 552+0 is designed for bending forces and uplift forces created by attachments such as the cross-over leg restraints. Downward forces are taken directly through bearing onto the foundation slab without imposing any bending or shear stresses on the base slab. The anchorage of the larger components is achieved by means of continuous steel connections through the liner plate into the foundation slab without creating stresses in the base slab.

Hand calculations are used for design since the loads are simple and the flat slab can easily be represented as a wide beam. Temperature and shrinkage steel is provided in the slab in areas where there are no applied loads and resulting stresses.

3.8.3.4.2 Reactor Vessel Cavity Wall

The reactor vessel cavity wall is represented as a space frame model for analysis purposes. The major loads include compartment pressure from postulated pipe breaks pressure, seismic forces and support loads from the reactor vessel and steam generator lower lateral supports. Other smaller loads are included for pipe supports and restraints.

3.8.3.4.3 Upper Reactor Cavity and Refueling Canal

The refueling canal floor and walls along with the upper reactor cavity walls are analyzed as a space finite element model. The design loads include seismic, internal and external

compartment pressures, and pipe support and restraint loads. Reactions from adjacent structural elements are included for the operating floor and the CRDM missile shields.

3.8.3.4.4 Crane Wall

The crane wall is analyzed as a space frame model. The model includes additional members and elements to represent the walls and slab that connect to the crane wall. Thus, the proper stiffness and interconnection with other elements is included. The applied loads include seismic forces, pressures from postulated pipe breaks, equipment loads, pipe support and restraint reactions, and reactions from adjacent structural elements.

The crane wall is divided into two sections for analysis. Both the upper and lower sections are modeled as space frames using STRUDL. For more details concerning governing loads and load combinations, critical design forces and the design of reinforcing bars, refer to Table 3-49.

3.8.3.4.5 Steam Generator Compartments

The removable steel shell portions of the steam generator enclosures are designed in accordance with Section III, Subsection NE of the 1974 ASME Code including addenda through the Summer of 1976. The steel dome is analyzed as a thin shell of revolution employing Kalnins' computer program for axisymmetric shells. The cylindrical steel shell portion of the enclosure is modeled as a plane frame for a typical horizontal section of the shell. The concrete portions of the enclosure are modeled using space frame members. The stiffness of the concrete walls is so much greater than the thin steel shell that no interaction is considered. The concrete displacements are included as boundary loads for the steel shell, and the steel shell reactions are included as loads on the concrete model.

The loads on the steel shell are from internal pressure due to a main steam line rupture or other postulated pipe break and also seismic forces. The forces on the concrete portion include pressure due to main steam line rupture or LOCA, seismic, and pipe support and restraint loads.

3.8.3.4.6 Pressurizer Compartment

The pressurizer compartment is designed for internal pressure due to pipe rupture, pressurizer support reactions, seismic forces, and jet impingement forces associated with postulated pipe ruptures. The compartment is modeled using space frame members and elements. The roof slab is included in the space frame model to represent the proper stiffness. An additional plate bending model with more detail is used, however, to design the roof slab.

3.8.3.4.7 Operating Floor

The operating floor is modeled using plate bending and stretching elements. Both in plane and out of plane forces are included. The in plane forces are due to support reactions from the steam generator upper lateral restraints. The major out of plane forces include differential pressure from a postulated pipe break and jet impingement from the associated pipe rupture. Other forces such as dead, live, seismic, and equipment and pipe support loads are also included.

Two separate analyses are performed using different element layouts and different computer programs. The analyses are conducted by two independent and separate groups of the Civil/Environmental Division of the Design Engineering Department. Each of the independent analyses are checked by qualified engineers within the respective groups and the comparison of results is reviewed for agreement by the Group Supervisors of each group and the Principal Engineer of the Structural Section.

One model is run using the STRUDL computer program and the other is run using the ELAS program. For comparison purposes, the two models are loaded with a unit pressure. The models are illustrated in Figure 3-280 and Figure 3-281. A comparison of the results is shown in Figure 3-282 through Figure 3-285. The close comparison between the programs assures the validity of the results.

3.8.3.4.8 Accumulator Floor

The accumulator floor at elevation 565+3 is modeled as a plane grid. Three separate models are used for the various similar panels of the floor. One model represents the portion of floor between wing walls enclosing the accumulators. A second model represents the portion of floor inside the fan compartments. The third model represents the portion of floor within the instrumentation room.

Each model includes the openings in the floor and spring supports to represent the structural steel columns supporting the perimeter. The design loads include pressures from a postulated pipe break, seismic forces, equipment and pipe support and restraint loads, dead, and live loads.

3.8.3.4.9 Ice Condenser Floor

The ice condenser floor at elevation 593+8 1/2 is subjected to regularly spaced uniform support loads from the lower support structure within the ice region. Therefore, a representative segment of the floor is modeled using a space frame model. The loads include pressure from a postulated pipe break, seismic forces, ice condenser lower support structure reactions, dead and live loads.

3.8.3.4.10 CRDM Missile Shield and Refueling Canal Gate

The CRDM missile shield beams and refueling canal gate sections are both simply supported one way spans. The analysis is therefore performed using hand calculations. Both are subjected to differential pressure due to a postulated pipe break and seismic forces. In addition, the CRDM missile shield beams are designed for dead, live, and internal missile loads. The missile loads are described in Section 3.5.1.2.

3.8.3.4.11 NSSS Support Systems

The design and analysis of the NSSS supports is fully described in Section 5.4.14.

3.8.3.4.12 Accumulator Wing Walls and Ice Condenser End Walls

These walls are modeled using plate bending elements. The major load is differential pressure from a postulated pipe break. Also included are equipment and pipe support reactions as well as seismic loads.

3.8.3.4.13 Computer Programs for the Structural Analysis

The following computer programs are employed in the analysis of Category I 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 Sections 3.7.2 and 3.8.2.4 for description of program.

2. 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.
3. For stresses and displacements of plates and frames subject to static and dynamic loads: The Structural Design Language (STRUDL) computer program, latest version and modification. Refer to the STRUDL manuals published by MIT and McDonald Douglas Automation Corporation for program description.
4. For linear equilibrium problems of general structures: The ELAS75 program. Refer to latest manuals published by Duke University, School of Engineering, December 1971.

3.8.3.5 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 3-32.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control, and construction techniques described in Section 3.8.1.6 also apply to the interior structure. In addition to the materials listed in Section 3.8.1.6.1, stainless steel plates (ASTM A240, Type 304) and stainless steel shapes (ASTM A276, Type 304) are also used. Concrete quality and placement techniques are in compliance with ACI 301, ACI 318, and Regulatory Guide 1.55 rather than ACI 349. Applicable materials, quality control, and construction techniques do comply with ASME Code, Section III, Division 1, Subsection NF, and ANSI N45.2.5 as appropriate.

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 Seismic Category I Structures

3.8.4.1 Description of the Structure

1. Auxiliary Building Including Control Room, Diesel Buildings, and Fuel Pools

   The Auxiliary Building is a poured-in-place, reinforced concrete structure as shown in Figure 1-8. The Auxiliary Building houses the Nuclear Steam Supply auxiliary equipment, electrical equipment control room, spent fuel pools, diesel generators, and related piping and cabling. The structure is designed to provide biological shielding and missile protection where applicable. Components of the Auxiliary Building are as follows:

   a. Diesel Generator Buildings

      There are two Diesel Generator Buildings, each housing two diesel generators. Each building is a reinforced concrete Category I seismic structure with plan dimensions of approximately 83 feet by 100 feet (See Figure 1-5). Each building is founded on rock and/or fill concrete with an eight foot thick reinforced concrete mat. The mat includes various equipment trenches and pits. The structural walls and roof slabs have a minimum concrete thickness of 36 inches. All of the Diesel Generator Buildings are located below grade except for the stairwells which project to a top elevation of 604+0. Each building is divided in half by a 36 inch reinforced concrete wall isolating diesel "A" from diesel "B".

The 14” diameter penetration for the 10” Nuclear Service Water System crossover header and the 6” diameter penetration for the 3” diesel generator room sump pump system (WN) piping are the only openings in the wall separating the A and B sides of the Diesel Generator Building. The 3 foot thick isolation wall is constructed of reinforced concrete.

b. Spent Fuel Buildings

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 foot 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 testing for leaks throughout the life of the station. The fuel transfer 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 120 feet in length, 21-1/2 feet in width and the maximum depth of water in the pool is approximately 40 feet. Each building is founded on rock or fill concrete with a four foot thick reinforced concrete mat. The thickness of the roof of the spent fuel pool is 24 inches. For further details of the pool, refer to Figure 1-4 through Figure 1-7.

The spent fuel buildings enclose the pools on all but the east end. The east end opens into the new fuel and cask handling area. This new fuel and cask handling area is enclosed by a Category I, reinforced concrete structure referred to as the New Fuel Building.

c. Control Complex

The Control Complex houses the battery room, cable room, and the control room. The Control Complex is a reinforced concrete Category I seismic structure. The control complex is founded on rock and/or fill concrete with a four foot thick structural mat. The roof slab on this complex is 24 inches thick. For further details on the location and dimensions of the control room, see Figure 1-6.

d. Outside Doghouse and UHI Buildings

The outside doghouse structures contain high pressure main steam and feedwater piping. The Upper Head Injection (UHI) tank and components for Unit 1 and the Boron Feed Tanks (BFT) for Unit 2 are housed in the UHI Building adjacent to the doghouse structure, separated by 36" thick reinforced concrete wall.

Each building is a reinforced concrete Category I structure. The outside doghouse structure is approximately 60 feet by 30 feet in plan dimension while the UHI enclosure is approximately 20 feet by 47 feet. Each building has a four foot thick reinforced concrete mat and is founded on rock and/or fill concrete. The structural walls are a minimum of 36 inches thick while the roof slab is 24 inches thick.

2. Nuclear Service Water System Structures

The Nuclear Service Water System Structures consist of a standby nuclear service water pond dam, intake structures, pump structure, and discharge structure.

The pump structure is a two-celled structure with each cell containing a NSW pump and related accessories. The pump structure is a reinforced concrete structure composed of foundation mat, walls, floor slab, and roof slab. See Figure 2-43 for a more detailed description of the structure.

The intake and discharge structures are reinforced concrete Category I Headwalls composed of Mat, Walls, and Roof Slab providing assured pipe openings designed for static and dynamic earth pressure forces. The openings in the intake structure are covered with metallic screens, and are sized for low level velocity to keep debris from entering the system. The structures will be completely submerged. See Figure 2-45 & Figure 2-46 for a more detailed description.

3.8.4.2 Applicable Codes, Standards, and Specifications

Category I structures will be designed in accordance with the codes and criteria as shown in Table 3-31.

3.8.4.3 Loads and Loading Combinations

Loads and combinations thereof used in the analysis and design of Category I Structures are tabulated in Table 3-32. The wind and tornado loads are described in Sections 3.3.1 and 3.3.2. The soil and water pressures are described in Section 2.5.4.10.4. Seismic loads are described in Section 3.7.

3.8.4.4 Design and Analysis Procedures

The Auxiliary Building is statically designed as a series of rigid plane frames (or in some instances, a three dimensional space frame) subjected to applicable loads summarized in Table 3-32. The structural Design Language (STRUDL) computer program is used to perform this static analysis.

The seismic effects are analyzed dynamically and are determined in accordance with the procedures described in Section 3.7.2.1.

Concrete design is in accordance with ACI 318, rather than ACI 349, and with Regulatory Guide 1.55 as appropriate.

The design of the reinforced concrete is performed in accordance with ACI 318-71, ultimate strength. Structural Steel is designed in accordance with AISC "Steel Construction Manual," Seventh Edition.

In both the static and dynamic analyses the vertical columns, walls, etc. are considered fixed at the base mat.

Masonry construction is designed and reinforced to remain functional under the above applicable loading conditions. Additional information concerning masonry construction was provided in W. O. Parker's letter of April 8, 1982 to H. R. Denton in response to SEB Action Item 34.

The arrangement of the fuel pool in relation to the fuel unloading area prohibits the cask from being moved over stored fuel.

3.8.4.5 Structural Acceptance Criteria

These are as outlined in Section 3.8.1.5.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

These are as outlined in Section 3.8.1.6.

3.8.4.7 Testing and Inservice Surveillance Requirements

These are as outlined in Section 3.8.1.7, with the following addition:

The spent fuel pool liner plate welds are inspected in accordance with:

1. Dye penetrant examination is performed in conformance to Appendix VIII of Section VIII of the ASME boiler and Pressure Vessel Code.
2. Upon completion of the dye penetrant tests, the weld seams are covered with a leak chase system and then pressure tested. All detected leaks are repaired and retested. This pressure testing of the leak chase system is repeated periodically through the life of the facility.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The foundation for the concrete Reactor Building is described in Section 3.8.1.1.

Foundation descriptions of other Category I structures are given in Section 3.8.4.1.

General descriptions of Category I foundations are given in Section 2.5.4.10.

Horizontal shears are discussed in Section 3.8.5.4.

3.8.5.2 Applicable Codes, Standards, and Specifications

See Table 3-31 for applicable codes and specifications. Standards are referenced in the applicable specifications.

3.8.5.3 Loads and Load Combinations

This information is given in Table 3-32.

3.8.5.4 Design and Analysis Procedures

The Reactor Building foundation is modeled as a circular plate on spring supports using the axisymmetric computer program used for the Reactor Building and containment vessel described in Sections 3.8.1 and 3.8.2 respectively. The reaction loads from the interior structures, containment vessel, and Reactor Building Wall are applied as ring loads using Fourier series representation.

The Reactor Building foundation is a 6 ft. thick reinforced concrete mat made of 5000psi concrete. The foundation is founded on rock with an allowable bearing capacity of 100,000psf. Anchorage to the rock is not required to provide an adequate factor of safety against overturning.

The critical moments and shears for foundation reinforcement design are the result of uplift loads from the reactor building walls, containment vessel, interior structures, and operating loads from equipment. Reinforcement design at all locations is based upon the maximum circumferential and radial moments obtained from analysis. The following are the maximum analysis results and the location where the results occur:

Analysis Results

Force	Magnitude	Location

Radial Moment	+1277 k-in/in	Crane wall
Radial Moment	-1432 k-in/in	Between crane wall and reactor vessel cavity
Circumferential Moment	612 k-in/in	All locations (a moment to 612 kin/in correspond. to min. reinfor. requirements).
Shear	+18.52 k/in	Reactor Vessel Cavity
Shear	-28.14 k/in	Crane Wall
Shear	7.19 k/in	All other locations (shear was less than concrete capacity).

Foundation mat reinforcement is designed for flexure and shear by the ultimate strength design method of ACI 318-71. Flexure reinforcement is designed at all locations using the maximum moments shown above. Shear reinforcement is provided where the allowable shear capacity of the concrete is exceeded. In addition to the flexure and shear reinforcement, dowels are provided at the junction of the foundation mat and the reactor building wall and the crane wall. These dowels are designed for the maximum moments that occur at the base of the wall. Consideration was given to development length into the foundation mat in order to fully develop the capacity of the dowels.

Refer to Figure 3-246 for reinforcement details at the reactor building wall, crane wall, and reactor vessel cavity.

Load combinations and factors of safety used for checking sliding and overturning are shown in Table 3-32. These calculations are relatively simple and are performed by hand once the sliding and overturning forces have been determined. The calculation of safety factors assumes no tension reaction.

All sections of ACI 318-71 which deal with ultimate strength design of reinforced concrete have been met in the reactor building foundation. This includes topics such as minimum concrete cover, minimum embedment length, and minimum reinforcement requirements.

The foundations for Category I structures other than the Reactor Building are analyzed as grid systems on elastic supports. The analysis is performed using Structural Design Language (STRUDL) computer program. Lateral loads are transferred to the foundation media by friction using a coefficient of friction of 1.0. Effects of settlement are given in Section 2.5.4.10.

3.8.5.5 Structural Acceptance Criteria

These are discussed in Sections 3.8.1.5, and 2.5.4.10.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

These are discussed in Section 3.8.1.6.

3.8.5.7 Testing and Inservice Inspection Requirements

There are no tests or inspections required in service for the foundations.

3.8.6 References

1. Kalinis, A., "Computer Programs for the Analysis of Axisymmetric Shells," 1971.

2. 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.
3. Control Data Corporation, "STARDYNE User Information Manual," Rev. C, April 1980.
4. McDonnell Douglas Automation Company, "ICES STRUDL User Manual," October 1981.
5. U.S. Atomic Energy Commission, "Regulatory Standard Review Plan," Section 3.8.2, NUREG-75/087, Rev. 0, 1975.
6. ASME Code Case N-284, "Metal Containment Shell Buckling Design Methods," August 1980.
7. Bushnell, D., "BOSOR4 Program for Stress, Buckling and Vibration of Complex Shells of Revolution," Structural Mechanics Software Series, Vol. 1, Perrone, N. and Pilkey, W. (editors), University of Virginia Press, 1974.
8. MacNeal-Schwendler Corporation, "MSC/NASTRAN User's Manual," Version 61, February 1981.
9. MARC Analysis Research Corporation, "MARC General Purpose Finite Element Program User Information Manual," Version J.1, June 1980.
10. Reissner, E., "A New Derivation of the Equations for the Deformations of Elastic Shells," American Journal of Mathematics, Vol. 63, 1941, pp. 177-184.
11. Kalnins, A., "Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads," Journal of Applied Mechanics, Volume 31, September 1964, pp. 467-476.
12. U.S. Nuclear Regulatory Commission, "Standard Review Plan," Section 3.8.2, NUREG-0800, Rev. 1, July 1981.

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3.9 Mechanical Systems and Components

Note:

This section of the FSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document (DBD) for this system/structure.

3.9.1 Special Topics for Mechanical Components

Fatigue Evaluation for License Renewal

Catawba Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 3.9.1 to assure that components are maintained within design limits. This requirement is managed by the Catawba *Thermal Fatigue Management Program.*

1.0 Thermal Fatigue Management Program

The four key actions of the *Thermal Fatigue Management Program* are:

1.1 Determining the Thermal Cycles to be Monitored and Their Character and Number of Allowed Occurrences: The set of transient events to be managed by the *Thermal Fatigue Management Program* is derived from the associated component information. Included are their thermal and pressure profile characteristics and the minimum of the numbers of occurrences used in the evaluations. As updates occur to associated component information such as analyzed conditions, operational practices, inservice inspection results, flaw growth analyses, or fatigue environmental effect modifications required for the extended period of operation (after 40 years), the set of transients and their limits may require revision.

1.2 Monitoring the Thermal Cycles Experienced: From continual monitoring of plant operating conditions, plant conditions that meet the definition of a transient cycle defined by this program are noted. Upon discovery of each transient cycle required to be documented by the program, the cycle count for that transient event is updated. For those events that are logged, the *Thermal Fatigue Management Program* specifies appropriate parameters such as minimum/maximum temperature limits and rates of temperature change that are assumed in the analysis. The logging process captures these values for review.

1.3 Comparison of Observed Events to Allowable Events: For the transients that have occurred since the previous assessment, two evaluations are performed to determine if parameters are within limits. The first evaluation compares the observed values for those parameters applicable to each transient to the limits described in the *Thermal Fatigue Management Program* (e.g. a maximum or minimum temperature limit). The second evaluation is a comparison to the allowable number of occurrences.

1.4 Corrective Action and Confirmation Process: Should the thermal and pressure profile for a specific transient be outside of the parameters defined for that transient set or should an allowable cycle count limit for a transient cycle set be approached or exceeded, this is identified to the appropriate engineering group(s) for resolution. The corrective action program is triggered immediately if profile values are exceeded by more than an immediately excusable amount. Similarly,

the corrective action program is triggered if the number of events is expected to exceed the thermal fatigue basis limits within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay within acceptable cycle count limits.

2.0 Future Modification to the TFMP for Environmentally Assisted Fatigue

The Thermal Fatigue Management Program will address the effects of the coolant environment on component fatigue life (environmentally assisted fatigue or EAF) by assessing the impact of the reactor coolant environment on a sample of critical locations selected from NUREG/CR-6260 and other locations expected to have high usage factors when considering environmentally assisted fatigue. The objective to meet in choosing locations will be to ensure by example that no plant location will have an EAF-adjusted CUF that exceeds 1.0 in actual operation.

The sample of critical components can be evaluated by applying the environmental correction factors to the existing ASME Code fatigue analyses and either (1) computing and tracking an EAF adjusted CUF against an allowable of 1.0 or (2) tracking the instances of transients identified in Paragraph 1.1 above against an EAF adjusted allowable number of transients.

Base formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels. Duke recognizes these formulas as the current methodology for determining such factors.

The exercise of the above procedure will be at a time prior to the end of the 40th year of each unit's operation. This lead time shall be sufficient to ensure that implementation of corrective actions will prevent the exceedance of 1.0 of EAF-adjusted CUF within the extended period of operation. No requirement exists that any resulting adjustments in allowables be applied prior to the end of the initial 40 years of operation. It is recognized that a discontinuity exists at the 40 year point in the need to apply this adjustment.

Duke may chose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with Duke.

References:

- Application to Renew the Operating Licenses of McGuire and Catawba, June 13, 2001 (existing UFSAR Supp Reference 18-1);
- *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (MRP-47)
- M.S. Tuckman (Duke) letter dated April 15, 2002, *Response to Requests for Additional Information in Support of the Staff Review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2,* Docket Nos. 50-369, 50-370, 50-413 and 50-414 (new UFSAR Supp Reference)

Leak-Before-Break Evaluation for License Renewal

Leak-before-break analyses evaluate postulated flaw growth in the primary loop piping of the Reactor Coolant System. These analyses consider the thermal aging of the cast austenitic stainless steel material of the piping, as well as the fatigue transients that drive the flaw growth over the operating life of the plant. Because all of the criteria contained in §54.3 are met, leak

before break is a TLAA for Catawba. The leak before break analyses have been determined to be acceptable for the period of extended operation.

3.9.1.1 Design Transients

This section provides a summary of the type and number of plant operational transients and related data which constitute the basis for analyzing and evaluating the cyclic behavior of ASME Code Class 1 and core support components, component supports, and reactor internals. These transients are constituted by the temperature, pressure, and flow transients resulting from the various operating conditions of the plant, and are based upon transients representative of operating conditions which prudently can be considered to occur during plant operation.

Those transients which were used for fatigue evaluations or fracture mechanics evaluations for the Reactor Coolant System and related Class 1 piping are represented in Table 3-50 along with the plant operating condition associated with each transient and the limiting allowable number of occurrences predicted or analyzed for each transient, considering ASME Section III and XI limits. In accordance with ASME Section III, faulted conditions were not included in fatigue evaluations.

Section 3.9.1.1 of revision 3 of Regulatory Guide 1.70 states "[p]rovide a complete list of transients to be used in the design and fatigue analysis of all ASME Code Class 1 and CS components, component supports, and reactor internals. The number of events for each transient should be included, along with assurance that the number of load and stress cycles per event is properly taken into account. All design transients that are contained in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary should be specified."

As stated above, Catawba Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 3.9.1 to assure that components are maintained within design limits.

Table 3-50 meets the RG 1.70 requirement to provide a listing of transients for which the applicable components have been qualified, with the following exceptions:

1. For practically purposes, local transient events that occur as sub-events of listed events are not tabulated in Table 3-50, but are defined in the details of the event descriptions of listed events as found and maintained in the applicable plant Engineering Records.
2. Emergency conditions are not part of the design basis for Catawba. Thus, no emergency conditions extraneously specified in any ASME Design Specifications are listed in Table 3-50.
3. As given in Column Note 2 of Table 3-50, some components/piping segments are individually qualified for faulted events which are not listed because the Unit as a whole is therefore not qualified for such.
4. In order to provide a definition of the envelope against which operation is to be compared, in compliance with Catawba Technical Specification 5.5.6, the occurrence quantity provided in Table 3-50 is the limiting allowable number of occurrences predicted or analyzed for each transient event, for all components and piping segments, considering ASME Section III and XI limits. Providing a complete tabulation of the design quantities (as required by RG 1.70), which varies from component to component, would be impractical. Such design information is however found and maintained in the applicable plant Engineering Records.

The following four operating conditions, as defined in Section III of the ASME B&PV Code, are considered in the design of the Reactor Coolant System (RCS), RCS component supports, and reactor internals.

1. Normal Conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, faulted or testing conditions.

2. Upset Conditions (Incidents of Moderate Frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the design specifications.

3. Faulted Conditions (Limiting Faults)

   Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

4. Testing Conditions

   Testing conditions are those tests in addition to the ten (10) hydrostatic or pneumatic tests permitted by ASME Section III including leak tests or subsequent hydrostatic tests.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of actual transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The following is a description of a selected subset of the primary system transients from Table 3-50 taken from Equipment Specifications for RCS components.

Normal Conditions

The following primary system transients are considered Normal Conditions:

1. Heatup and Cooldown at 100°F per hour
2. Unit Loading and Unloading at 5 Percent of Full Power/per minute
3. Step Load Increase and Decrease of 10 Percent of Full Power
4. Large Step Load Decrease with Steam Dump

5. Steady State Fluctuations

1. Heatup and Cooldown at 100°F Per Hour

   The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F per hour. The expected normal rates are <50°F per hour for heatup and <80°F per hour for cooldown.

   For these cases, the heatup occurs from ambient to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

   a. Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

   b. Slower initial heatup rates when using pump energy only.

   c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

   The controlling number of such complete heatup and cooldown operations, 200 each, corresponds to five such occurrences per year for a 40 year plant design life.

2. Unit Loading and Unloading at 5 Percent of Full Power per Minute

   The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15 percent load and full load. This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the Reactor Control System. The controlling number of loading and unloading operations, 13,200, is comparable to one loading operation per day which yields 14,600 such operations during a 40 year design life.

3. Step Load Increase and Decrease of 10 Percent of Full Power

   The ±10 percent step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The Reactor Control System is designed to restore plant equilibrium without reactor trip following a ±10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load, the power range for automatic reactor control. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

   Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average

temperature setpoint change is made as a function of turbine-generator load as determined by the first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The controlling number of each operation is 2000 times or 50 per year for a 40 year plant design life.

4. Large Step Load Decrease With Steam Dump

   This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator safety valves.

   The controlling number of occurrences of this transient is 200 times or 5 per year for a 40 year plant design life.

5. Steady State Fluctuations

   The reactor coolant average temperature, for purposes of design, is assumed to increase or decrease a maximum of 6°F in one minute. The temperature changes are assumed to be around the programmed value of Tavg, (Tavg±3°F). The corresponding reactor coolant average pressure is assumed to vary accordingly.

Upset Conditions

The following primary system transients are considered Upset Conditions.

1. Loss of load (without immediate reactor trip)
2. Loss of power
3. Partial loss of flow
4. Reactor trip from full power
5. Inadvertent reactor coolant system depressurization
6. Reactor trip with cooldown and safety injection actuation
7. Inadvertent Auxiliary Spray
8. Operating basis earthquake.

1. Loss of Load (Without Immediate Reactor Trip)

   This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under Upset Conditions. The reactor eventually trips as a consequence of a high

pressurizer level trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

The controlling number of occurrences of this transient is 80 times or 2 times per year for a 40 year plant design life.

2. Loss of Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 100 percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are de-energized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 60°F (Unit 1) or 32°F (Unit 2), from the Auxiliary Feedwater System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

The controlling number of occurrences of this transient is 40 times or 1 per year for a 40 year plant design life.

3. Partial Loss of Flow

This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The controlling number of occurrences of this transient is 80 times or 2 times per year for a 40 year plant design life.

4. Reactor Trip From Full Power

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

The controlling number of occurrences of this transient is 230 times or approximately 5 times per year over a 40 year plant life.

5. Inadvertent Reactor Coolant System Depressurization

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. Actuation of a single pressurizer safety valve.

b. Inadvertent opening of one pressurizer power operated relief valve due either to equipment malfunction or operator error.

c. Malfunction of a single pressurizer pressure controller causing one power operated relief valve and two pressurizer spray valves to open.

d. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error.

e. Inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as an "umbrella" case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

As the pressure drops, the reactor is tripped and SI is actuated. Also, the SI accumulators are actuated when the pressure drops below 650 ± 50 psi. The controlling number of occurrences is 20 during the plant life.

6. Reactor Trip with Cooldown and Inadvertent Safety Injection Actuation

This transient is caused by excessive feedwater flow following a normal reactor trip. For design purposes, it is assumed that the cooldown continues until SI is actuated and the feedwater flow is isolated. The SI charging pump repressurizes the system. The controlling occurrence limit is 10 times during the plant life.

7. Inadvertent Auxiliary Spray

The inadvertent auxiliary spray transient will occur if the auxiliary spray valve is opened inadvertently during normal operation of the plant. This will introduce cold water into the pressurizer with a very sharp pressure decrease as result.

The temperature of the auxiliary spray water is depending upon the performance of the Regenerative Heat Exchanger. The most conservative case is when the letdown stream is shut off and the charging fluid enters the pressurizer unheated. Therefore, for design purposes, the temperature of the spray water is assumed to be 70°F. The spray flow rate is assumed to be 200 gpm. It is furthermore assumed that the auxiliary spray will, if actuated, continue for five minutes until it is shut off.

The pressure decreases rapidly to the low pressure reactor trip point. At this pressure the pressurizer low pressure reactor trip is assumed to be actuated; this accentuates the pressure decrease until the pressure is finally limited to the hot leg saturation pressure. At five minutes spray is stopped and all the pressurizer heaters return the pressure to 2250 psia. Again if the pressurizer heaters were not in operation the pressure would remain at the value reached in five minutes.

For design purposes it is assumed that no temperature changes in the Reactor Coolant System with the exception of the pressurizer occur as a result of initiation of auxiliary spray.

The controlling number of occurrences of this transient during the life of the plant is specified as 10.

8. Operating Basis Earthquake

The mechanical stresses resulting from the operating basis earthquake (OBE) are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The

origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis.

Faulted Conditions

The following primary system transients are considered Faulted Conditions. Each of the following accidents should be evaluated for one occurrence:

1. Reactor Coolant Pipe Break (Loss of Coolant Accident)
2. Large Steam Line Break
3. Safe Shutdown Earthquake

1. Reactor Coolant Pipe Break (Large Loss of Coolant Accident or Pipe Rupture)

   Following a postulated rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the SIS is actuated to introduce water at a minimum temperature of 70°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

2. Large Steam Line Break

   This transient is based on the complete severance of the largest steam line. The following conservative assumptions were made:

   a. The reactor is initially in a hot, zero-power condition.

   b. The steam line break results in immediate reactor trip and in actuation of the Safety Injection Systems.

   c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transients.

   d. The high head safety injection pumps of the SIS repressurize the RCS to the extent of their shutoff head.

3. Safe Shutdown Earthquake

   The mechanical dynamic or static equivalent loads due to the vibratory motion of the Safe Shutdown Earthquake are considered on a component basis.

Test Conditions

The following primary system transients under Test Conditions are discussed:

1. Turbine Roll Test
2. Primary Side Hydrostatic Test
3. Primary Side Leakage Test

1. Turbine Roll Test

   This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour design rate.

The controlling number of such test cycles is 10 times, to be performed at the beginning of plant operating life prior to irradiation. This transient occurs before plant startup and the number of cycles is therefore independent of other operating transients.

2. Primary Side Hydrostatic Test

   The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3106 psig (1.25 times design pressure), or 2690 psig, depending on the component. In this test, the RCS is pressurized coincident with steam generator secondary side pressure of 0 psig. The controlling quantity for the RCS is 5 cycles of these hydrostatic tests, normally performed prior to plant startup. The number of cycles is independent of other operating transients.

   Additional hydrostatic tests will be performed to meet the inservice inspection requirements of ASME Section XI. A total of four such tests is normally expected. The increase in the fatigue usage factor caused by these tests is easily covered by the conservative controlling number (50) of primary side leakage tests.

3. Primary Side Leakage Test

   Subsequent to each time the primary system has been opened, a leakage test would be performed. During this test the primary system pressure is, for design purposes, raised to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

   For design purposes, the controlling quantity is 50 cycles of this test during the life of the plant.

   This test is no longer performed separate from normal heatup and pressurization.

3.9.1.2 Computer Programs Used in Analysis

3.9.1.2.1 Components and Equipment

In the qualification of specific components or equipment provided to Duke, vendors may use computer methods of analysis. These components and equipment must be qualified in accordance with the functional and structural criteria set forth in the applicable Duke specification.

3.9.1.2.2 Piping Systems

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED.

Static and dynamic analyses of Catawba piping systems are performed using the computer program SUPERPIPE which is owned and maintained by Impell Corporation (formally EDS Nuclear Inc.), 455 North Wiget Lane, Walnut Creek, California 94598-0591.

SUPERPIPE is a general purpose piping program which performs comprehensive structural analyses of linear elastic piping systems for dead weight, thermal expansion, seismic spectra or time history, arbitrary force time history and other loading conditions. Analyses are performed to ASME requirements for Class 1, 2 and 3 systems.

A piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The location of lumped masses is chosen to accurately

represent the dynamic characteristics of the system for a dynamic analysis, and to adequately represent the weight distribution of the system for dead load analysis. Static or dynamic equilibrium equations are formulated using the direct stiffness method, in which element stiffness matrices are formed according to virtual work principles and assembled to form a global stiffness matrix for the system, relating external forces and moment to joint displacements and rotations. Appropriate stiffness modifications for curved components are included. Diagonal mass and damping matrices are assumed.

Static equilibrium equations are solved using Gaussian reduction techniques on the global stiffness matrix. For dynamic problems, the equilibrium equations may be solved using either step-by-step direct integration of the coupled equations of motion, or by first calculating natural frequencies and mode shapes and transforming the system into a set of uncoupled equations of motion. For seismic analysis of piping systems the latter approach is typically used in the dynamic analysis technique known as the response spectrum mode superposition method. In this technique, the earthquake excitation is characterized by acceleration response spectra, and the total response of the system is evaluated as a combination of the individual responses of the significant natural modes of vibration of the system. Natural frequencies and mode shapes are calculated using the determinant search technique. The method of combination of model responses can be selected from any one of those specified in Regulatory Guide 1.92. Earthquakes acting in all three directions simultaneously may be computed.

SUPERPIPE has been thoroughly verified for a comprehensive set of sample problems. This has included bench marking by EDS against the ASME Sample Problems 1 and 6 contained in ASME publication "Pressure Vessel and Piping 1972, Computer Program Verification", and against a Class 1 sample problem contained in ASME publication "Sample Analysis of a Piping System, Class 1 Nuclear", 1972. Extensive bench marking has also been performed by EDS against the programs, PISOL1A and PISOL3A which are well recognized and utilized throughout the industry. Additionally, the program has been bench marked by EDS against the programs such as NUPIPE, ADLPIPE, PIPESD and EDSGAP. SUPERPIPE has been used on a number of domestic and foreign plants. These include South Texas, McGuire 1, and San Onofre 1 and 2 (United States); Tihange 2 (Belgium); Kernkraftwerk Kruemmel, and Kernkraftwerk Phillipsburg (Germany); Kerndraftwerk Iran (Iran); Almaraz, Cofrentes and Valdecaballeros (Spain); and, Leibstadt (Switzerland).

Additional descriptions of computer codes used and their verification is provided in Table 3-56.

3.9.1.2.2.1 ADLPIPE for Buried Diesel Generator Supply and Return Piping

ADLPIPE is a computer program for the structural analysis of complex piping systems subjected to static and dynamic loads.

This PC based program isused to analyze piping systems addressed in Reference 32 approved by the NRC in References 33 and 34.

The analyses were performed by a vendor, J.D. Stevenson and Associates, under their internal Quality Assurance program. Additional description and its verification is provided in Table 3-120.

3.9.1.2.3 Programs Supplied by NSSS Vendor

3.9.1.2.3.1 Westinghouse Scope

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. These are described and verified in References 1 and 2.

1. WESTDYN-7 - static and dynamic analysis of redundant piping systems.
2. FIXFM - time-history displacement response of three-dimensional structures.
3. WESDYN-2 - piping system stress analysis from time history displacement data.
4. STHRUST - hydraulic loads on loop components from blowdown information.
5. WECAN - finite element structural analysis.
6. DARI - WOSTAS - dynamic transient response analysis of reactor vessel and internals.
7. SATAN IV - Space time dependent analysis of loss of coolant accident that treats all phases of blowdown loads.

Duke Fuel Assembly Compatibility Evaluation for the supply of 17 x 17 Westinghouse Robust Fuel Assemblies

Computer code "DARI-WOSTAS" was not used, instead, the WECAN (References 1 and 2) computer code, which is already listed, was used for the dynamic transient response analysis of the reactor vessel and internals.

3.9.1.2.3.2 Computer Codes Supplied by Vendor for Steam Generator Replacement (Unit 1)

The following computer programs have been used in static and dynamic analyses to determine mechanical loads, stresses and deformations of Seismic Category I components and equipment. These codes are described and verified in References 17 and 18 in Section 3.9.7.

1. BWSPAN - static and dynamic analysis of linear piping systems.
2. CRAFT2 - LOCA and mass/energy release analysis

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis methods have been used for the Catawba project.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

This section describes the faulted condition load combinations and analysis methods for reactor coolant system piping, components, and supports. As noted in Section 3.6, pipe breaks in the primary loop RCS piping have been eliminated from consideration in certain aspects of the plant design, as defined in Reference 16. However, reactor coolant system piping (including Class 1 branch lines), primary components, and their supports have been designed and analyzed for the faulted condition SRSS load combination of SSE and LOCA (postulated pipe break in main RCS piping). This approach provides considerable margin in the plant design. The following sections describe the faulted condition analyses including the analysis methods used for LOCA.

3.9.1.4.1 Loading Conditions

The structural stress analyses performed on the reactor coolant system consider the loadings specified as shown in Table 3-51. These loads result from thermal expansion, pressure, dead weight, Operating Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE), design basis loss of coolant accident, and plant operational thermal and pressure transients.

3.9.1.4.2 Analysis of the Reactor Coolant Loop

The reactor coolant loop piping is evaluated in accordance with the criteria of ASME III, NB-3650 and Appendix F. The loads included in the evaluation result from the SSE, deadweight, pressure, and LOCA loadings (loop hydraulic forces, asymmetric subcompartment pressurization forces, and reactor vessel motion).

The loads used in the analysis of the reactor coolant loop piping are described in detail below.

Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at change in direction or flow area. These loads apply only to Unit 2.

Dead Weight

A dead weight analysis is performed to meet Code requirements by applying a 1.0g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

Seismic

The forcing functions for the reactor coolant loop seismic piping analyses are derived from dynamic response analyses of the containment building subjected to seismic ground motion. Input is in the form of floor response spectrum curves at the highest elevation where the RCS is attached to the containment building.

For the OBE and SSE seismic analyses, see Section 3.7.1.3 for the damping values used in the reactor coolant loop/supports system analysis.

In the response spectrum method of analysis, the total response loading obtained from the seismic analysis consists of two parts; the inertia response loading of the piping system and the differential anchor movement loading. Two sets of seismic moments are required to perform an ASME Code analysis. The first set includes only the moments resulting from inertia effects and these moments are used in the resultant moment (M_i) value for Equations 9 and 13 of NB-3650. The second set includes the moments resulting from seismic anchor motions and are used in Equations 10 and 11 of NB-3650. Differential anchor movement is discussed in Section 3.7.

Loss of Coolant Accident

Blowdown loads are developed in the broken and unbroken reactor coolant loops as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loops. Structural consideration of dynamic effects of postulated pipe breaks requires

postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6.

Broken loop and unbroken loop time history dynamic analyses are performed for these postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6.

Transients

The Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are tabulated in Section 3.9.1.1.

The vertical thermal growth of the reactor pressure vessel nozzle centerlines is considered in the thermal analysis to account for equipment nozzle displacement as an external movement.

The hot moduli of elasticity E, the coefficient of thermal expansion at the metal temperature α, the external movements transmitted to the piping due to vessel growth, and the temperature rise above the ambient temperature ΔT, define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the Reactor Coolant System, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions.

3.9.1.4.3 Reactor Coolant Loop Models and Methods

3.9.1.4.3.1 Westinghouse Scope (Unit 2)

The analytical methods used in obtaining the solution consists of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectra method for seismic dynamic analysis, and time history integration method for the loss of coolant accident dynamic analysis. The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affect the system. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The reactor coolant loop/supports system model, constructed for the WESTDYN-7 computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3-287 shows an isometric line schematic of this mathematical model. The steam generator and reactor coolant pump vertical and lateral support members are described in Section 5.4.14.

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus elasticity E, the coefficient of thermal expansion , the average temperature change from ambient temperature ΔT, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrices which define restraint characteristics of the supports. The reactor pressure

vessel centerline is represented by a fixed boundary in the system mathematical model. The vertical thermal growth of the reactor vessel nozzle centerline is represented by a fixed boundary in the system mathematical mode. The vertical thermal growth of the reactor vessel nozzle centerline is considered in the construction of the model.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for deadweight, thermal, and general pressure loading conditions are obtained by using the WESTDYN-7 computer program. The derivation of the hydraulic loads for the loss of coolant accident analysis of the loop is covered in Section 3.6.2.

Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. The effect of the equipment motion on the reactor coolant loop/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is represented by three discrete masses. The lower mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The middle mass is located at the steam generator upper support elevation and the third mass is located at the top of the steam generator.

The reactor coolant pump is represented by a two discrete mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles. The upper mass is located near the center of gravity of the motor.

The reactor vessel and core internals are represented by approximately 13 discrete masses. The masses are lumped at various locations along the length of the vessel and along the length of the representation of the core internals.

The steam generator upper and lower lateral supports are inactive (i.e. do not provide support) during plant heatup, cooldown and normal plant operating conditions. These restraints become active (i.e. provide support only) when the plant is at power. They are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The response spectra method employs the lumped mass technique, linear elastic properties, and the principle of modal superposition. Floor response spectra are generated for two perpendicular horizontal directions and the vertical direction. The floor response spectra are applied along each horizontal axis simultaneously with the vertical axis.

From the mathematical description of the system, the overall stiffness matrix K is developed from the individual element stiffness matrices using the transfer matrix method. After deleting the rows and columns representing rigid restraints, the stiffness matrix is revised to obtain a

reduced stiffness matrix (KR associated with mass degrees of freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determined. The modal participation factor matrix is computed and combined with the appropriate response spectra value to give the modal amplitude for each mode. The total modal amplitude is obtained by computing the absolute sum of the contributions for each direction.

The modal amplitudes are then converted to displacements in the global coordinate system and applied to the corresponding mass point. From these data the forces, moments, deflections, rotations, support reactions and piping stresses are calculated for all significant modes.

The total seismic response is computed by combining the contributions of the significant modes by using the methods described in Section 3.7.

Loss of Coolant Accident

The mathematical model used in the static analyses is modified for the loss of coolant accident analyses to represent the severance of the reactor coolant loop piping at the postulated break location. Modifications include addition of the mass characteristic of the piping and equipment. To obtain the proper dynamic solution, two masses, each containing six dynamic degrees of freedom and located on each side of the break, are included in the mathematical model. The natural frequencies and eigenvectors are determined from this broken loop model.

The dynamic structural solution for the loss of coolant accident is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they are modelled as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple non-linear elements can be applied at the same node, if necessary.

The time-history solution is performed in subprogram FIXFM3. The input to this subprogram consists of the natural frequencies, normal modes, applied forces and nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN-7 program. To properly simulate the release of the strain energy in the pipe, the internal forces in the system at the postulated break location due to the initial steady-state hydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response to the system from the static equilibrium position). The time history displacement solution of all dynamic degrees of freedom is obtained using subprogram FIXFM3 and employing 4 percent critical damping. (Damping values - Reference 14)

The time-history displacements of the FIXFM subprogram are used as input to a program WESDYN 2 to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses.

The displacements obtained from the reactor vessel analysis (considering the effect of asymmetric cavity pressure, reactor vessel internals reaction, and loop mechanical forces) are statically applied as an imposed boundary condition on the broken and unbroken loops to determine the deflection and stresses in the piping and loads on the supports.

The loads resulting from the loop analysis are combined absolutely with the loads resulting from the applied reactor vessel displacements.

The asymmetric pressure loads resulting from a postulated pipe rupture and pressure buildup in the loop compartments are applied to the same RCL/supports system model discussed previously. The response of the system is obtained for the various external pressure loading cases from which support member forces and piping stresses are calculated. The loop piping stresses resulting from the external pressure loading are added to the piping, stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The support loads are computed by multiplying the support stiffness matrix and the displacement vector at the support point. The support loads are used in the evaluation of the supports.

Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the ASME Code into three parts, a uniform, a linear, and a non-linear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the non-linear portion causes a skin stress.

The transients as defined in Section 3.9.1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program is used to solve the thermal transient problem. The pipe is represented by at least fifty elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 3-288.

The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed for all time steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H}\int_O^H T(X,t)dX$$

The range of temperature between the largest and smallest value of T_A is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

The thermal moment about the mid-thickness of the wall caused by the temperature distribution through the wall is equal to:

$$M = E\alpha\int_O^H (X - \frac{H}{2})T(X,t)dX$$

The equivalent thermal moment produced by the linear thermal gradient as shown in Figure 3-288 about the mid-wall thickness is equal to:

$$M_L = E\alpha\frac{\Delta T_1}{12}H^2$$

Equating M_L and M, the solution for ΔT_1 as a function of time is:

$$\Delta T_1(t) = \frac{12}{H^a} \int_0^H \left(X - \frac{H}{2}\right) T(X,t)\, dX$$

The maximum nonlinear thermal gradient, ΔT_2, will occur on the inside surface and can be determined as the difference between the actual metal temperature on this surface and half of the average linear thermal gradient plus the average temperature.

$$\Delta T_{2I}(t) = T(O,t) - T_A(t) - \frac{\Delta T_1(t)}{2}$$

Load Set Generation

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects axial thermal gradients at a given time in each transient. The method of load set generation is based on Reference 3. The through-wall thermal effects are functions of time and can be subdivided into four parts:

1. Average temperature (T_A) is the average through-wall temperature of the pipe which contributes to general expansion loads.
2. Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).
3. Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.
4. The axial thermal gradient, defined by the discontinuity temperature (T_A - T_B) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least two load sets: they represent the maximum and minimum stress states during each transient.

The construction of the max (min) load set is accomplished by calculating the stresses for the thermal effects (ΔT_1, ΔT_2, and T_A - T_B) on a time history basis, and choosing the ΔT_1, ΔT_2, and T_A - T_B effects at the time yielding the max (min) thermal moments are also included as well as the max (min) pressure seen during the transients. The procedure outlined yields the most accurate results for each transient using one-dimensional heat transfer, and produces at least twice as many load sets as transients for each point in the piping system.

As a result of the normal mode spectral technique employed in the seismic analysis the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus insuring the most conservative combination of seismic loads are used in the stress evaluation.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors (K_e) and cumulative usage factors, U, are calculated. The WESTDYN-7 program is used to perform this analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3650. Since it is impossible to predict the order of occurrence of the transients over a forty-year life, the most conservative and restrictive combination of transients is considered. Therefore, it is conservatively assumed that the transient can occur in any sequence.

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.1.4.3.2 Reactor Coolant Loop Reanalysis for Steam Generator Replacement (Unit 1)

The Unit 1 reactor coolant system was reanalyzed to take into account the effects of the Babcock and Wilcox International (BWI) replacement steam generators. This analysis was performed by B&W Nuclear Technologies (BWNT) of Lynchburg, VA. The reanalysis was defined to be a parametric analysis where the response of the reactor coolant system with the replacement steam generators was compared to the system response with the original steam generators.

The finite element method was used in obtaining the solution for the static, seismic dynamic, and loss of coolant accident dynamic analyses. The response spectrum method was used to generate the results for the seismic analysis while the time history method was used to generate the results for the loss of coolant accident analysis. The integrated reactor coolant system model consisted of the piping, components, and component supports. The component supports are described in Section 5.4.1.4. The NSSS model was coupled to the containment interior structure model (Figure 3-217) to facilitate the input of the design ground response spectra for the seismic analysis. The NSSS model (including the containment interior structure) is shown in Figures 3-301 through 3-304.

The NSSS system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, component supports and the containment interior structure. The model geometry is based on the reactor coolant loop piping layout and equipment drawings. All joint coordinates and piping/equipment element lengths are obtained from the drawings. The physical and material properties of each element are obtained from the drawings, specifications, and the ASME code. The piping and replacement steam generators were modeled using consistent mass model elements; the reactor vessel, reactor coolant pumps, and the containment interior structure were modeled with discrete lumped masses. The component supports are represented by spring elements, with the exception of the steam generator and reactor coolant pump columns which are modeled as beam elements.

The static solutions for the deadweight and thermal loading conditions are calculated using the stiffness method of analysis. The stiffness matrix and load vector are assembled and solved using the BWNT proprietary computer code BWSPAN. The piping deflections, internal forces, and stresses were determined at each node point. Support loads were calculated for each active component support. Operating pressure loads (see Section 3.9.1.4.2) were not considered in the Unit 1 NSSS reanalysis. BWNT evaluated these so-called "steady state momentum" loads and determined that they could be deemed negligible.

As described in Section 3.9.1.4.3.1, the steam generator upper and lower lateral supports are inactive during plant heatup, cooldown, and normal plant operating conditions. This is also true of the reactor coolant pump lateral restraints. (The reactor vessel vertical and lateral restraints are subject to thermal loads as are the steam generator and reactor coolant pump columns.) These lateral restraints become active (only) when the plant is at power. All dynamic analyses are performed for the full power condition.

The seismic analysis of the NSSS model uses the response spectrum method, which is based on the principal of modal superposition. Damping values are specified in Section 3.7.1.3. Nonproportional damping that reflects the material composition of the model (as described in Section III, Appendix N of the ASME Code) is used in the analysis. The plant design ground response spectra (Figure 3-213) are used in the seismic analysis as the NSSS model is coupled to the containment interior structure model. The spectra are applied along each horizontal axis simultaneously with the vertical axis (i.e. a pair of two-dimensional spectrum analyses are performed). The seismic acceleration in the vertical direction is two-thirds of that applied in the horizontal direction. The modal responses are combined as described in Section 3.7.3.7. The

three components of earthquake motion are then combined as described in Section 3.7.2.6(2) in order to obtain the maximum forces, moments, deflections, rotations, support reactions, and piping stresses.

The NSSS piping model was also evaluated for loss of coolant accident loads. The double ended guillotine breaks on the main reactor coolant loop piping were eliminated through the application of leak-before-break (see Section 3.6.2.1.1.1). Break loads were only evaluated for primary system branch line (pressurizer surge line, accumulator line, and residual heat removal line) and secondary system line (main steam line and main feedwater line) pipe breaks. The piping finite element model was subjected to a time history load for each defined pipe break case. The force time histories were calculated using the CRAFT2 computer code (see Reference 18 in Section 3.9.7). Support participation was reviewed and those supports at which the specified gap did not close were considered to be inactive. Asymmetric pressure load time histories were applied concurrently with the pipe rupture force time histories. All piping forces, moments, displacements, and stresses as well as components support loads were calculated using the BWSPAN computer code.

The thermal transients associated with the replacement steam generators were compared to the original system design thermal transients. The original design thermal transients were found to bound the thermal transients for the replacement steam generators. Stresses due to the thermal transients were not evaluated further.

3.9.1.4.4 Analysis of Primary Components

3.9.1.4.4.1 Westinghouse Scope

Equipment which serves as part of the pressure boundary in the reactor coolant loop include the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is ANS Safety Class 1 and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3-51. The equipment is analyzed for 1) the normal loads of dead-weight, pressure and thermal, 2) mechanical transients of OBE, SSE, and pipe ruptures, including the effects of asymmetric pressurization, and 3) pressure and temperature transients outlined in Section 3.9.1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analyses, a set of loads are determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the Unit 2 steam generators and Units 1 and 2 pressurizers are performed using 2 percent damping for the OBE and 4 percent damping for the SSE (Reference 14). Seismic analysis for the Unit 1 steam generators are performed using Regulatory guide 1.61 damping. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump

internals is performed using the damping for bolted steel structures, that is, 2 percent for the OBE and 5 percent for the SSE (0.5 percent for OBE and 1.0 percent for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump since the main flange, motor stand, and motor are all bolted assemblies (See Section 5.4). The reactor pressure vessel is seismically qualified by the reactor vessel vendor in accordance with the ASME Code. The loadings used in this analysis are supplied by Westinghouse and are based on loads generated by a dynamic systems analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the line connect to the primary system piping are orificed to a 3/8 inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

3.9.1.4.4.2 Primary Component Reanalysis for Steam Generator Replacement (Unit 1)

The results of the reactor coolant loop analysis with replacement steam generators were used to evaluate the qualification of the reactor vessel, reactor coolant pumps, and replacement steam generators. The nozzle loads calculated in the NSSS reanalysis were compared to those used for the original equipment design to show that all primary components were acceptable. Other loading conditions for the existing primary components did not change.

The replacement steam generators were designed for the loading combinations shown in Table 3-51. The replacement generators were designed to withstand normal loads (deadweight, pressure, and thermal), mechanical transients (OBE, SSE, and pipe rupture, including the effects of asymmetric pressure loads), and pressure and temperature transients associated with normal and abnormal plant operating conditions (see Section 3.9.1.1). Umbrella nozzle loads for the replacement steam generators were supplied for all loading conditions. Actual nozzle loads from the reactor coolant loop reanalysis were compared to the umbrella loads to insure that the design was acceptable. The seismic analysis for the replacement steam generators was performed using damping values of 2% for OBE and 3% for SSE per Regulatory Guide 1.61. The replacement steam generators were designed to meet all requirements of Section III, Subsection NB of the ASME Boiler and Pressure Vessel Code.

3.9.1.4.5 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss of Coolant Accident

The structural analysis of the reactor vessel and internals for a postulated LOCA considers simultaneous application of the time-history loads resulting from the reactor coolant loop mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization. The vessel is restrained by reactor vessel support pads and shoes beneath four of the reactor vessel nozzles, and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

Reference 16 documents the transmittal of a Westinghouse report that justifies the elimination of RCS primary loop breaks from certain design considerations. Table 3-20 summarizes the postulated break locations to be considered for the main coolant loop. The following break locations and sizes have been analyzed:

Pressurized Surge Line

On the RCS hot leg, a 98.31 in^2 break was postulated at the pressurizer surge line branch connection safe-end juncture with the surge line.

RHR Lines

Also on the RCS hot leg, an 86.6 in^2 break was postulated at the RHR line branch connection safe-end juncture with the RHR line for the Catawba loop location nearest the vessel.

Accumulator Line

On the RCS cold leg, a 60 in^2 break was postulated at the accumulator line branch connection safe-end juncture with the accumulator line for the Catawba loop location nearest the vessel.

By considering these breaks (which bound smaller postulated branch line breaks), the most severe reactor vessel loads are determined.

Duke Fuel Assembly Compatibility Evaluation for the supply of 17 x 17 Westinghouse Robust Fuel Assemblies.

Changes in fuel assembly properties generally impact the performance of the reactor pressure vessel and its internals under all modes of operation. It is, therefore, important that with a change of fuel the mechanical response of the reactor pressure vessel and its internals be evaluated. The transition of Catawba 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.

3.9.1.4.5.1 Loading Conditions

Following a postulated pipe rupture the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of three phenomena: (1) reactor coolant loop mechanical loads, (2) reactor cavity pressurization forces and (3) reactor internal hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. The loop mechanical forces which are released at the broken nozzle are applied to the vessel in the RPV blowdown analysis.

Reactor cavity pressurization forces arise for the pipe breaks at the vessel nozzles from the steam and water which is released into the reactor cavity through the annulus around the broken pipe. The reactor cavity is pressurized asymmetrically with higher pressure on the side of the broken pipe resulting in horizontal forces applied to the reactor vessel. Smaller vertical forces arising from pressure on the bottom of the vessel and the vessel flanges are also applied to the reactor vessel. The cavity pressure analysis is described in Section 6.2.

The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a cold leg break, the depressurization wave path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down and around the downcomer annulus and up through the fuel.

In the case of a hot leg break the wave passes through the RPV outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus, for a hot leg break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the cold leg break. For both the hot and cold leg breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been

referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized region. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708, (Reference 13).

3.9.1.4.5.2 Reactor Vessel and Internals Modeling

The reactor vessel is restrained by two mechanisms: (1) the four attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and (2) four reactor vessel supports, two beneath reactor vessel inlet nozzles and two beneath reactor vessel outlet nozzles. The reactor vessel supports are described in Section 5.4.14. The support shoe provides restraint in the horizontal directions and for downward reactor vessel motion.

The reactor vessel model consists of two non-linear elastic models connected at a common node. One model represents the dynamic vertical characteristics of the vessel and its internals, and the other model represents the translational and rotational characteristics of the structure. These two models are combined in the DARI-WOSTAS code (Reference 1) to represent motion of the reactor vessel and its internals in the plane of the vessel centerline and the broken pipe centerline.

The model for horizontal motion is shown in Figure 3-289. Each node has one translational and one rotational degree of freedom in the vertical plane containing the centerline of the nozzle attached to the broken pipe and the center-line of the vessel. A combination of beam elements and concentrated masses are used to represent the components including the vessel, core barrel, neutron panels, fuel assemblies, and upper support columns. Connections between the various components are either pin-pin rigid links, translational impact springs with damping or rotational springs.

<u>Duke Fuel Assembly Compatibility Evaluation for the supply of 17x17 Westinghouse Robust Fuel Assemblies</u>

<u>Mathematical Model of the Reactor Pressure Vessel (RPV)</u>

The mathematical model of the RPV is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the Catawba reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN (Reference 1 and 2) computer code. The WECAN model replaces the old methodology which was performed using the DARI WOSTAS code. Shown in Figure 3-305 is 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 3-306, represents the reactor vessel shell and associated components. The reactor vessel is restrained by four reactor vessel supports (situated beneath alternate nozzles) and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical impact element. The attached piping is represented by a stiffness matrix.

The second submodel, shown in Figure 3-307, 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 3-308, 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 P+ 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:

$$M_f = \frac{M_H \mid -(M_1 + M_H)}{-(M_1 + M_H) \mid (M_1 + M_2 + M_H)}$$

where

$M_1 =$ $\pi R^2{}_i L\rho$

$M_2 =$ $\pi R^2{}_o L\rho$

$M_H =$ hydrodynamic mass = $\frac{R_o^2 + R_i^2}{R_o^2 - R_i^2}\ \rho\pi R^2{}_i L$

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 participate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artifical 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 3-309.

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 equation (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

1. Three-dimensional elastic pipe
2. Three-dimensional mass with rotary inertia
3. Three-dimensional beam
4. Three-dimensional linear spring
5. Concentric impact element
6. Linear impact element
7. 6 x 6 stiffness matrix
8. 18 Card stiffness matrix
9. 18 Card mass matrix
10. Three-dimensional friction element

3.9.1.4.5.3 Analytical Methods

The time history effects of the cavity pressurization loads, internals loads and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

<u>Duke Fuel Assembly Compatibility Evaluation for the supply of 17x17 Westinghouse Robust Fuel Assemblies</u>

The finite element models shown in Figures 3-305 through 3-308 were used to perform the LOCA analysis. Catawba takes credit for leak-before-break (LBB). The limiting breaks to be considered are the branch line breaks which consists of (a) accumulator line, (b) pressurizer surge line, and (c) 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 Catawba reactor pressure vessel system consist 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.

<u>RPV Internal Hydraulic Loads</u>

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 a break as for the 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, Reference 13, 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.

Reactor Coolant Loop Mechanical Loads

The reactor coolant loop mechanical loads are applied to the RVP 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 hydraulic 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, i.e., accumulator, pressurizer surge, and RHR lines, the restraints on the main coolant piping would eliminate any force to the reactor vessel caused by a break in the auxiliary line breaks.

3.9.1.4.5.4 Results of the Analysis

As described, the reactor vessel and internals were analyzed for three postulated break locations. Table 3-54 summarizes the displacements and rotations of and about a point representing the intersection of the centerline of the nozzle attached to the leg in which the break was postulated to occur and the vertical centerline of the reactor vessel. Positive vertical displacement is up and positive horizontal displacement is away from and along the centerline of the vessel nozzle in the loop in which the break was postulated to occur. These displacements were calculated using an assumed break opening area for the postulated pipe ruptures at the branch lines, as described in Section 3.9.1.4.5. These areas are estimated prior to performing the analysis. Following the reactor coolant system structural analysis, the relative motions of the broken pipe ends are obtained from the reactor coolant loop blowdown analysis. The actual break opening area is then verified to be less than the estimated area used in the analysis and assures that the analysis is conservative.

The maximum loads induced in the vessel supports due to the postulated pipe break are given in Table 3-55. These loads are per vessel support and are applied at the vessel nozzle pad. It is conservatively assumed that the maximum horizontal and vertical loads occur simultaneously and on the same support, even though the time history results show that these loads occur neither simultaneously nor on the same support.

The largest vertical loads are produced on the support, opposite the broken nozzle. The largest horizontal loads are produced on the supports which are perpendicular to the broken nozzle horizontal centerline.

Duke Fuel Assembly Compatibility Evaluation for the supply of 17x17 Westinghouse Robust Fuel Assemblies

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 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 reactor vessel.

The loads described in the previous sections were applied to the WECAN model of the reactor pressure vessel system shown in Figure 3-305 through Figure 3-308 and the input to the analysis was specifically applicable to Catawba. A detailed discussion of the analyses performed for typical fuel assemblies is contained in Reference 4 of Section 3.7.

The transition of Catawba to Westinghouse 17x17 Robust fuel with IFMs will not adversely impact the response of the reactor internals system and components due to LOCA excitations.

3.9.1.4.6 Stress Criteria for Class 1 Components and Component Supports

All Class 1 components and supports are designed and analyzed for the design, normal, and upset conditions to the rules and requirements of the ASME Code Section III. The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with supplementary options outlined below:

1. Elastic System Analysis and Component Inelastic analysis

   This is an acceptable method of evaluation for Faulted Conditions if primary stress limits for components are taken as greater of 0.70 S_u or S_y + 1/3 (S_u - S_y) for membrane stress and greater of 0.70 S_{ut} or S_y + 1/3 (S_{ut} - S_y) for membrane plus-bending stress, where material properties are taken at appropriate temperature.

   If plastic component analysis is used with elastic system analysis or with plastic system analysis, the deformations and displacements of the individual system members will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

2. Elastic/Inelastic System Analysis and Component/Test Load Method

   The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

   If the component/test load method is used with elastic or plastic system analysis, the deformations and displacements of the individual component members taken from the test load method data at the loads resulting from the system analysis will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

   A list of seismic Category I equipment and the method of qualification used is provided in Table 3-1, Table 3-2, and Table 3-4.

Loading combinations and allowable stresses for ASME III Class 1 components and supports are given in Table 3-51, Table 3-52 and Table 3-53. For Faulted condition evaluations, the effects of the safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) are combined using the square root of the sum of the squares (SRSS) method. Justification for this method of load combination is contained in References 4 and 5. The responses to other loading combinations defined in Table 3-51 are combined using the absolute sum method.

3.9.1.4.7 Balance-of-Plant Components, Piping and Supports

Seismic category I piping other than NSSS is analyzed for the faulted condition utilizing elastically-determined stresses compared against allowables provided in Table F-1322.2-1 of Appendix F of the ASME Code Section III. This is in accordance with applicable sections of the ASME Code or ANSI B31.1 as appropriate. Load combinations and allowable stresses for faulted and other plant conditions are discussed in Section 3.9.3.

Dynamic seismic analysis for the SSE is performed on this piping utilizing the model combination method in accordance with USNRC Regulatory Guide 1.92.

All seismic Category 1 supports are designed and analyzed for the Normal, Upset, Faulted and Test Conditions. The stress limits for normal and upset conditions are as presented in ASME III Subsection NF and Subsection NA Appendix XVII for the portion of the support within the NF boundary. The stress limit for the faulted load combination is as specified in Subsection NF with the exception that to avoid column buckling in compression, for members subject to local instability associated with compression flange buckling in flexural members and web buckling in plate guides, the allowable stress has been limited to 2/3 of the critical buckling stress. For support design there is no inelastic analysis. Temperature effects for material properties are considered. For the portion of the support not within the NF boundary and for supports for B31.1 piping, stress limits are as provided in MSC-SP58 or the AISC Manual.

For integral attachments to the pressure boundary the rules of ASME Section III, Subsection NB, NC, ND are used as applicable.

3.9.2 Dynamic Testing and Analysis

HISTORICAL INFORMATION NOT REQUIRED TO BE UPDATED.

3.9.2.1 System Operational Test Program

3.9.2.1.1 System Vibration Testing

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. Details of the tests are given in Figure 14-2.

Duke Class A, B, C, and F systems satisfy the criteria of Regulatory Guide 1.68, Revision 2, Appendix A, 5.o.o for systems to be included in the vibration test program. Systems which will be subject to steady state vibration testing are identified in Table 3-84. 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 testing is not performed on piping with nominal size 1 in. or less, with the exception of the Reactor Coolant System instrumentation lines (including pressurizer level and reactor*

vessel level) which are specifically included in the test program. The consequences of the failure of small line does not justify the expense of designing them to meet the vibration requirements.

2. *Vibration testing is not performed on piping containing gases, rather than liquids, because the relatively small forces exerted by flowing gases preclude the development of excessive vibration. High flow velocity steam lines are an exception and will be tested.*

3. *Vibration testing is not performed on piping systems which have no flow, or have 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.*

The acceptance criteria for piping vibration is that the maximum measured amplitude shall not induce a stress in the piping greater than one-half the endurance limit corresponding to 10^6 cycles as defined in Section III of the ASME Boiler and Pressure Vessel Code, 1974, Summer 1974 Addenda.

In the steady state, vibration testing of piping, the systems will be placed in the normal operating mode. Qualified personnel shall perform a visual inspection of the systems, noting locations of maximum vibration. These locations will be used for the measurement of the pipe vibration. Data collected, with suitable instrumentation, will be compared with acceptance criteria based on the piping material (carbon or stainless steel). If an unfiltered vibration reading exceeds the acceptance criteria, a spectrum analyzer will be used to obtain a spectrum plot of the vibration at that point. The 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.

The piping systems listed in Table 3-84 will be subjected to routine transients, valve closures, pump starts, etc., during system functional testing. Inspections will be carried out by qualified personnel after the transient event to verify the occurrence of any excessive piping motion. Excessive motion will be evidenced by induced damage to piping supports, loosened hangers, out-of-range snubbers, damaged spring cans, etc. If excessive movement or vibration is indicated, an evaluation will be done by Design Engineering and corrective action taken as necessary.

Transient vibration testing will be done on systems as listed in Table 3-85. A graded approach is used in testing.

Acceptance criteria for transient vibration testing is based on conservative design of piping and restraints. If excessive movement or vibration is indicated, an evaluation will be done by Design Engineering and corrective action taken as necessary.

Reactor internals preoperational tests and reactor internals vibration monitoring are presented in Section 3.9.1.4.1.

3.9.2.1.2 System Thermal Expansion Testing

Duke Class A, B, C, and F portions of piping systems with operating temperatures above 200°F will be visually inspected to ensure that piping and components are unrestricted from expanding. Systems to be included in the thermal expansion testing are listed in Table 3-86. Snubber and Spring Can movements will be verified within the design specified range. Details of these tests are presented in Figure 14-2. Reactor Coolant System expansion and restraint test is a special test with a separate abstract.

The acceptance criteria for thermal expansion testing is based on piping systems being designed to allow expansion without interference from piping supports or other piping systems.

If any piping or component is found to be restricted from expanding, Design Engineering will evaluate the problem and recommend support/restraint adjustment or modification as necessary to correct the problem. Acceptable response would then be verified.

When practical, the above tests will be performed during the preoperational testing phase. Those tests that cannot be performed as a part of the preoperational testing phase because of required plant conditions will be performed as a part of the initial startup and power escalation phase.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

3.9.2.2.1 Seismic Qualification Testing of Safety-Related Mechanical Equipment Furnished by Westinghouse

The operability of Category I mechanical equipment must be demonstrated if the equipment is determined to be active, i.e., mechanical operation is relied on to perform a safety function. The operability of active Class 2 and 3 pumps, active Class 1, 2, or 3 valves, and their respective drives, operators and vital auxiliary equipment will be shown by satisfying the criteria given in Section 3.9.3.2. Other active mechanical equipment will be shown operable by either testing, analysis or a combination of testing and analysis. The operability programs implemented on this other active equipment will be similar to the program described in Section 3.9.3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10 for electrical equipment will be used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive seismic Category I equipment will be shown to have structural integrity during all plant conditions in one of the following manners: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment, or 2) by test showing that the equipment retains its structural integrity under the simulated test environment.

3.9.2.2.2 Seismic Qualification of Safety-Related Mechanical Equipment Furnished by Duke

Seismic Category I mechanical equipment, including supports, is qualified to meet the performance requirements for structural integrity and functional operability during and following an earthquake of magnitude up to and including the Safe Shutdown Earthquake (SSE).

The actual seismic input motions are characterized by one of the following methods:

1. Response spectra
2. Power spectral density function
3. Time history

One of these methods is used to define the earthquake motion along each of three mutually orthogonal axes. These specified motions are representative of the actual motions determined by structural seismic analyses at the equipment mounting locations.

Seismic Category I mechanical equipment, including supports, will be qualified for both structural integrity and functional operability by one of the following methods.

1. Mathematical analysis
2. Testing of the equipment under simulated seismic conditions
3. Combined test and mathematical analyses (i.e., partial testing supplemented by mathematical analysis).

Determination of the method to be used is based upon the practicability of the method for the type, size, shape, and complexity of the equipment to be qualified. Mathematical analysis is used when the performance of equipment can be accurately predicted by analysis of the structural integrity. Testing is used when it is the most practical method of qualification or when analytical techniques are inadequate. Combined testing and mathematical analysis is used when it is impractical to qualify the equipment by either analysis or testing alone due to type, size, shape or complexity.

When analyses and/or testing procedures include evaluation of the equipment in the operating mode, including faulted conditions, design limits and loading combinations are established satisfying the requirements of Regulatory Guide 1.48 as discussed in Sections 3.9.3.2 and 3.9.3.3.

3.9.2.2.2.1 Qualification by Analysis

This method is used for equipment, including its supports, which does not lend itself to testing and which can be properly modeled and mathematically analyzed to obtain its response during the seismic event.

The mathematical analysis method shall consist of, but not necessarily be limited to, the following:

1. The equipment is mathematically modeled as a multi-degree-of-freedom, lumped-mass system with massless interconnections.
2. The natural frequencies and mode shapes of the equipment and supports are determined.
3. If all natural frequencies of the assembled equipment are greater than 33 Hz, the equipment is analyzed statically. In this static analysis, the seismic forces on each component of the equipment are obtained by concentrating its mass at its center of gravity and multiplying it by the appropriate maximum floor accelerations. The seismic stress is added to the equipment's operating stresses, and a determination is made of the adequacy of the strength of the equipment and its supports. The resulting deformations are combined with those associated with the equipment's operating deformations and a determination is made of the operability of the equipment under such deformations.
4. If any natural frequencies of the assembled equipment are less than 33 Hz, the equipment is analyzed dynamically. The equipment is modeled with sufficient mass points to ensure adequate representation. The resulting system is analyzed using the response spectra modal analysis technique or a time history (modal or step-by-step) analysis technique. A stress analysis is then performed using the inertia forces or equivalent static loads obtained from the dynamic analysis for each mode. If the modal analysis is used, the total seismic stress is normally obtained by taking the square root of the sum of the squares of the individual modal stresses, provided adjacent modal frequencies differ by more than 10 percent, otherwise the modal response is obtained by the absolute sum technique.
5. The analysis includes consideration of two mutually orthogonal components of horizontal seismic motion occurring simultaneously with the vertical motion. The horizontal directions are in most cases the principal axes of the equipment, but different axes are chosen if they result in a greater seismic response.
6. The equipment stresses induced from the earthquake loads as obtained above, shall be combined with stresses from other loads in accordance with Section 3.9.3.2.

3.9.2.2.2.2 Qualification by Testing

If testing is utilized to prove compliance with the seismic criteria, the requirements set below are met:

1. Wherever practical, the equipment to be tested is production equipment identical to that specified.
2. Similar equipment may be tested, provided the vendor demonstrates that any differences in design or construction are such that the dynamic responses are not significantly different from that of the specified equipment.
3. As an alternate to testing, the vendor may submit for approval the procedures and results of tests previously conducted on equipment, meeting requirements 1 or 2 above, which has been subjected to dynamic loads equal to or greater than that required for this plant.
4. Vibration testing of the specified equipment or component is performed on a shake table or other test equipment capable of applying the required vibratory motion to the specimen under test. Means are provided for controlling the direction of the applied motion and for setting and measuring the frequency and amplitude within the range prescribed in the specified input motion.

   When practicable, the equipment under test is attached to the mounting fixtures of the test machine using the actual equipment supports.

   Alternately, the equipment under test may be attached to the mounting fixtures of the test machine in a manner which conservatively simulates that which will be used in the installation.

   If the assembly is too large to mount on the shake table, the test may be run by soft mounting the assembly to the floor using flexible supports with resonance outside the frequency band of the test and rigidly connecting the base of the assembly to the vibration generator.
5. The preferred excitation is random motion waveform for the required input motion when the required response spectrum is broadband in nature. Testing with other waveform characteristics (frequency content) may be used depending on the test in question. The adequacy of testing with waveforms other than random motion will be justified for each type of equipment and its actual installation arrangement.

   Single frequency testing may be used to determine (or verify) the resonant modes and damping of the equipment. Single frequency input, such as sine beats, may be applicable provided one of the following conditions is met:

   a. For equipment supported at the higher elevations of the structure, structural filtering effects significantly reduce the frequency band width of the seismic disturbance which can be accurately represented by a dominant single frequency motion.
   b. The predicted response of the equipment is adequately represented by one mode.

      The input has sufficient intensity and duration to excite all modes to the point that the test response spectra envelops the required response spectra for the equipment. The test consists of application of sine beats with test frequency equal to the equipment single frequency of interest and the peak acceleration equal to the applicable maximum floor response acceleration (required response spectrum acceleration value at 33 Hz or greater). The beat frequency is determined to insure sufficient severity so that the test response spectrum envelops the required response spectrum. Repeated beats, with

pauses between beats to eliminate superposition of beat responses, are used to represent low cycle fatigue effects over the duration of the applicable earthquake.

c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra envelope the corresponding response spectra of the individual modes. The test at each frequency consists of the application of sine beats of peak acceleration equal to the maximum floor response (response spectrum acceleration value at 33 Hz or greater). The sine beats consists of a sinusoid at the frequency, 5 beats used, with a pause between the beats such that no significant superposition of motion results. There are 10 cycles per beat.

6. Single axis testing can be used when justified by one of the following criteria. Otherwise multi-axis testing must be used.

   a. The equipment is not sensitive to cross-coupling of the response to motion along each of the orthogonal axes.

   b. If the failure mode is along a single axis, the input acceleration level is increased to conservatively account for the amount of coupling.

   The input motion is applied simultaneously to the vertical axis and a horizontal axis parallel to the front surface of the equipment when the equipment is in its normal mounting position. Independent random inputs are preferred and, if used, the test is performed in two steps with the equipment rotated 90° in the horizontal plane for the second step. If inphase inputs are used (such as with single frequency tests), four tests should be run. First, with the inputs in phase; next, with one input 180° out of phase; next, with the equipment rotated 90° horizontally, and the inputs in phase; and, finally with the same equipment orientation but with one input 180° out of phase.

7. An exploratory vibration test may be run on the equipment or component along each axis independently to determine the resonant frequencies. The test is run in the form of a continuous-sweep-frequency search using a low magnitude, steady-state, sinusoidal input with a minimum frequency range of 1 to 33 Hz.

   The level of input is chosen to provide a usable signal to noise ratio on vibration sensing equipment. For equipment which are suspected to exhibit non-linear behavior, excitation levels and sweep rates are adjusted to ascertain high level resonant frequencies in addition to the low level resonances.

8. To ensure operability of all components, if it is not practicable to test complex assemblies in a complete operational mode, it is acceptable to test such equipment in an operative mode with the actual or simulated devices installed. The response accelerations at these locations are monitored. The individual devices are separately tested while operating, and the acceptable accelerations are then verified.

9. In order to demonstrate functional capability of the equipment under test, analyses are made to determine the necessity of simulating the various modes of operation that the equipment may be experiencing before, during, and after the earthquake event. Testing is conducted with sufficient instrumentation and monitoring devices to evaluate performance before, during and after the test.

10. Vendors will utilize documented testing procedures detailing the method of testing to be used, parameters to be measured, input motion proposed, and all other information required to adequately perform the test specified.

11. The in-situ application of vibratory devices to superimpose the seismic vibratory loading on complex active pumps or valves for operability testing is acceptable when application is justified.
12. If previously stated criteria for single axis and single frequency testing is not met, multi-axis and multi-frequency testing is used. For multi-axis, single frequency testing the minimum criteria is biaxial testing with simultaneous inputs in a horizontal axis and the vertical axis. Testing will be performed with inputs to each horizontal axis with simultaneous vertical axis input separately. Two tests are run for each horizontal-vertical axis combined input set-up: First with inputs inphase; next with one input 180 degrees out-of-phase. It is required that the test response spectra envelop the required response spectra along each axis. Single frequency inputs consist of sine beats derived as described in Paragraph E of this section. For multi-axis, multi-frequency testing the minimum criteria is also biaxial testing with simultaneous inputs along each horizontal axis and the vertical axis in two separate test set-ups as described above. Inputs for each axis are synthesized time histories developed from the required response spectra for that axis. Beforehand it is demonstrated that the actual test machine motion is equal or greater than the required motion. Duration of the input excitations are sufficiently long to simulate the effects of the required seismic event. Since inputs of this nature are random, no input phasing is required during test. It is required that the test response spectra envelop the required response spectra along each axis.

3.9.2.2.2.3 Equipment Supports

Supports for all Category I mechanical equipment are seismically qualified by analysis and/or testing. The requirements equivalent to those of Sections 3.9.2.2.1 and 3.9.2.2.2 are fulfilled, together with the following criteria.

1. Analytical Methods

   All effects of the supported equipment are conservatively included in the analysis.

2. Test Procedures

   When practicable, the actual equipment mount will be tested. Alternately, the test mount will conservatively represent the actual service mount.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations can not be practically performed independently of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions. The effects of these forcing functions have been

studied from tests performed on models and prototype plants, as well as component tests (References 6, 7, and 8).

The Indian Point 2 plant has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant instrumentation program and the Sequoyah I plant instrumentation program provide prototype data applicable to Catawba (References 6 and 8).

The Catawba plant is similar to Indian Point 2; and the only significant differences are the modifications resulting from the use of I7 x I7 optimized fuel, replacement of the annular thermal shield with neutron shielding pads, and the change to the UHI inverted top hat support structures configuration. These differences are addressed below.

1. I7 x I7 Fuel

   The only structural changes in the internals resulting from the design change from the I5 x I5 to the I7 x I7 optimized fuel assembly are the guide tube and control rod drive line. The new I7 x I7 guide tubes are stronger and more rigid, hence they are less susceptible to flow induced vibration. The optimized fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation is expected from the I5 x I5 fuel assembly vibration characteristics.

2. Neutron Shielding Pads Lower Internals

   The primary cause of core barrel excitation is flow turbulence generated at the inlet nozzle and in the downcomer annulus (Reference 8). The vibration levels due to core barrel excitation for Trojan and Catawba both having neutron shielding pads, are expected to be similar. Since Catawba has a slightly higher flow rate than Trojan, vibration levels due to the core barrel excitation are expected to be somewhat greater than those of Trojan. Scale model test results (Reference 7) and results from Trojan (Reference 6) show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stress and large safety margins were measured at Indian Point 2 (thermal shield configuration) lead to the conclusion that stresses less than those of Indian Point 2 will result on the Catawba internals with the attendant large safety margins.

3. UHI Inverted Top Hat Upper Support Configuration

   The components of the upper internals are excited by turbulent forces due to axial and cross flows in the upper plenum (Reference 8) and pump speed related components. Sequoyah and Catawba have the same upper internals configuration; therefore, the general vibration behavior is not changed. Data on upper internals vibration have been obtained during hot functional testing at Sequoyah 1. A preliminary report on analysis of the data has been submitted. A final report including measurements with the core in place is in preparation. Reduction of the data and the post hot functional inspection results provide assurance of the design adequacy. The increased flow rate of Catawba with respect to Sequoyah is reflected in upper internals vibrations primarily as a change in fluid velocity. The vibration of the upper internals due to flow turbulence is approximately proportional to the product of density and velocity squared (Reference 8). This product is approximately 5% higher in Catawba than Sequoyah 1. By applying the 5% increase in the quantity v^2 (i.e. density times velocity squared) to the high factors of safety deduced from the Sequoyah 1 data, the minimum factor of safety for Catawba upper internals is 1.8, (Reference 14). 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 startup testing of Sequoyah 1. These data indicate lower vibration levels (and consequently higher factors of safety) than those deduced from hot functional data.

The original test and analysis of the four-loop configuration is augmented by (References 6, 7, and 8) to cover the effects of successive hardware modifications.

HISTORICAL INFORMATION NOT REQUIRED TO BE UPDATED.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the Catawba reactor internals design configuration is well characterized, as was discussed in Section 3.9.2.3, it is not considered necessary to conduct instrumented tests of the Catawba plant hardware. The prototype plant for Catawba is Indian Point Unit 2. The IPP-2 plant was fully instrumented and tested during hot functional and pre-operational testing. Additionally, available test data on the Trojan 1 and Sequoyah 1 plants together with the prototype IPP-2 results can be used to characterize vibrational characteristics of 4-loop internals. The significant differences between Catawba and IPP-2 internals are the replacement of the annular thermal shield with neutron panels, modifications resulting from the use of 17x17 fuel and the change to UHI-style inverted top hat upper internals.

The upper internals of Catawba and that of the tested Sequoyah 1 unit are similar with the UHI-style inverted top hat configuration. The upper internals adequacy of Sequoyah 1 has been established by the plant tests and supplemented with the scale model tests of similar configurations. The results of testing at Sequoyah 1 show that components are excited by flow induced and pump related excitations. Analyses of the data indicate that the instrumented components have adequate factors of safety and that the vibration behavior is well characterized. A specific comparison of the Sequoyah upper head characteristics with the Catawba plant is provided in Section 3.9.2.3. This information demonstrates that appropriate safety margins exist for the upper internals. In summary, structural adequacy and vibratory behavior of the Catawba upper internals configuration has been established by testing at the Sequoyah 1 plant.

Scale model tests indicate significantly lower vibrational levels for internals with neutron panels than for internals with annular thermal shields. Test results from Trojan 1 (neutron panels similar to Catawba) show lower vibration levels than on IPP-2. The primary source of excitation of the core barrel is flow turbulence generated at the inlet nozzles and in the downcomer. Since both Catawba and Trojan 1 have neutron panels, the vibration levels are similar. The coolant inlet temperature and flow rate of Catawba are slightly higher than Trojan 1. Scale model tests show that the core barrel vibration levels vary as the velocity raised to a small power. The differences in fluid density and flow rate result in an approximately 5.8% higher core barrel vibrations for Catawba when compared with Trojan 1. This correlation and the fact that the scale model tests and plant tests show that vibration levels are lower with neutron panels than annular thermal shields leads to the conclusion that stresses less than or approximately equal to IPP-2 will result on the Catawba internals.

Fuel assembly masses and stiffnesses remain relatively unchanged, and so no significant change in internals vibration is expected.

The recommendations of Regulatory Guide 1.20, Position C.3, are satisfied by conducting the confirmatory pre- and post- hot functional examination for integrity. This examination includes in excess of 30 features (illustrated in Figure 3-291) with special emphasis on the following areas:

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.

4. Those other locations on the reactor internal components which are similar to those which are examined on the prototype designs.

5. The inside of the vessel will be 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.

A particularly close inspection will be made on the following items or areas using a 5X or 10X magnifying glass or other appropriate inspection.

1. Lower Internals

   a. Upper barrel to flange girth weld.

   b. Upper barrel to lower barrel girth weld.

   c. Upper core plate aligning pin. Examine bearing surfaces for 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 pads, screw locking devices and dowel pin lock welds. Examine the interface surfaces for evidence of tightness. Check 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 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 bearing surfaces for shadow marks, burnishing, buffing or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. Subsequent to the hot functional testing, the bearing surfaces of the key and keyway will show burnishing, buffing or shadow marks which indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.

2. Upper Internals

   a. Thermocouple conduits, clamps and couplings.

   b. Guide tube, support column, flow column, and thermocouple assembly locking devices.

   c. Support column and thermocouple conduit assembly clamp welds.

   d. Upper core plate alignment inserts. Examine bearing surfaces for shadow marks, burnishing, buffing or scoring. Check the locking devices for integrity of lockwelds.

   e. Thermocouple conduit fitting locking tab and clamp welds.

f. Guide tube enclosure 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 will be 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 $l0^7$ cycles on the main structural elements of the internals. In addition there will be some operating time with only one, two and three pumps operating.

Pre- and post- hot functional inspection results serve to confirm that the internals are well behaved. When no signs of abnormal wear and harmful vibrations are detected and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operation.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

1. Loss of coolant accident (LOCA). Both cold leg and hot leg breaks are considered.
2. Safe shutdown earthquake (SSE).

Maximum stresses for SSE and LOCA are obtained and combined.

Maximum stress intensities are compared to allowable stresses for each of the above conditions. Elastic analysis is used to obtain the response of the structure and the stress analysis on each component is performed according to ASME Code approved techniques. For faulted conditions, stresses are compared to meet the Code allowable design limits provided in Table F-1322.2-1 of Appendix F of the ASME Code.

The criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core 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 for the internals are concerned with the deflections and stability of the parts in addition to stress criteria to assure integrity of the components.

For the critical internal structures, maximum allowable deflections, based on functional performance criteria, are listed in Table 3-87. The basic operational or functional criterion to be met for the reactor internals is that the plant shall be shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits following the design basis accident. The reactor internals structures have been conservatively designed to withstand the stress and be within deflection limits originating from a LOCA (full double-ended RCS primary loop pipe break) even though such pipe breaks are no longer considered for dynamic effects, according to Reference 16.

The reactor internals structures have been conservatively designed to withstand the stress and be within deflection limits originating from a LOCA (full double-ended RCS primary loop pipe break) even though such pipe breaks are no longer considered for dynamic effects, according to Reference 16.

Reactor Internals Analysis

Analysis of the reactor internals for 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 mechanical 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 blowdown digital computer program 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 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 of channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, as well as some effect of the water-solid interaction, are considered.

The blowdown code evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. Each reactor component for which 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 calculation, 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 has been performed using the following assumptions:

1. The analysis considers the effect of hydroelasticity.
2. 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.
3. The model described 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 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 a 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 internals 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 simultaneous seismic event with the intensity of the SSE is postulated with the loss of coolant accident, the combined effect of the maximum stresses for each case is considered. In general, the loading imposed by the earthquake is small compared to the blowdown loading. The seismic analysis of the reactor internals is discussed in Section 3.7.3.

A summary of the mechanical analysis is presented in the following paragraphs. Reference 9 provides further details of the method used in the reactor internals blowdown analysis.

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. In order to obtain an accurate simulation of the reactor internals response, the effects of internal damping, clearances between various internals, 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 10) which computes the response of the multi-mass model when excited by the time dependent hydraulic 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 were analyzed.

Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown and are analyzed to determine their response to this excitation. The core barrel, guide tubes, and upper support columns analyses are discussed in the following paragraphs.

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.

2. The shell is treated as simply supported.

During cold 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 core barrel is analyzed as a shell with two variable sections to model the core barrel flange and core barrel.
2. The barrel with the core is analyzed as a beam elastically supported at the top and at the lower radial support and the dynamic response is obtained.

Guide Tubes - The dynamic loads on rod cluster 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. The guide tubes in closest proximity to the ruptured outlet nozzle are the most severely loaded. The transverse guide tube forces during a blowdown decrease with increasing distance from the ruptured nozzle location.

A detailed structural analysis of the rod cluster control guide tubes is performed to establish the equivalent cross section properties and elastic end support conditions. An analytical model is verified by subjecting the control rod cluster guide tube to a concentrated force applied at the midpoint of the lower guide tube. In addition, the analytical model has been previously verified through numerous dynamic and static tests performed on the l7 x l7 guide tube design.

The response of the guide tubes to the transient loading from blowdown is found by representing the guide tube as an equivalent single degree of freedom system and assuming the slope of the time dependent load to be a step function with constant slope front end.

Upper Support Columns - Upper support columns located close to the broken nozzle during hot leg break will be 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 and appropriate stress risers for the various sections.

Results of Reactor Internals Analysis

Maximum stresses due to the safe shutdown earthquake (vertical and horizontal components) and a loss of coolant accident (hot leg or cold leg break) were obtained and combined. All core support structure components were found to be within acceptable stress and deflection limits for a loss of coolant accident occurring simultaneously with the safe shutdown earthquake; the stresses and deflections which would result following a faulted condition are less than those which would adversely affect the integrity of the core support structures. For the transverse excitation, it is shown that the barrel does not buckle during a hot leg break and that it meets the allowable stress limits during all specified transients.

Also, the natural and applied frequencies are such that resonance problems will not occur.

The results obtained from linear analyses indicate that the relative displacement between the components will close the gaps and consequently the structures will impinge on each other. Linear analysis will not provide information about the impact forces generated when components impinge on each other; however, in some instances, linear approximations can, and are applied prior to and after gap closure. The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, between fuel assemblies and baffle plates, and

between the control rods and their guide paths were considered in the analysis using both linear approximations and non-linear techniques. Both static and dynamic stress intensities are within acceptable limits.

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. For the guide tubes deflected above the no loss of function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be 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.

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

As stated in Section 3.9.2.3, it is not considered necessary to conduct instrumented tests of the Catawba reactor vessel internals. Adequacy of these internals will be verified by use of the Sequoyah and Trojan results.

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures

The ASME Code Class components are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

Detailed discussion of ASME Code Class 1 components is provided in Sections 3.9.1 and 5.4.

For core support structures, design loading conditions are given in Section 3.9.5.2. The design loading combinations for the ASME Code case support structures are given in Table 3-89. It is to be noted that the reactor internals of this plant are not "stamped" and no specific stress report is required. Nonetheless, the internals are designed to meet the intent of the ASME Code.

In general, for reactor internals components and for core support structures the criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core 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.

For the loss of coolant plus the safe shutdown earthquake condition, deflections of critical internal structures are limited. In a hypothesized downward vertical displacement of the internals, energy absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

The following mechanical functional performance criteria apply:

1. Following the design basis accident, the functional criterion to be met for the reactor internals is that the plant shall be shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of critical components must be kept sufficiently small to allow core cooling.

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 a subcritical state. Therefore, the main requirement is to assure effectiveness of the

Emergency Core Cooling System. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

3. 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.

Methods of analysis and testing for core support structures are discussed in Sections 3.9.2.3, 3.9.2.5, and 3.9.2.6. Stress limits are given in Section 3.9.5.4. Deformation criteria is given in Sections 3.9.2.5 and 3.9.5.3.

Evaluation for License Renewal

Catawba 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.

3.9.3.1 Loading Combinations Design Transients, and Stress Limits (For ASME Code Class 2 and 3 Components)

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9.3.1.1 Design Loading Combinations and Design Stress Limits for Westinghouse Equipment

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3-88. The design loading combinations are categorized with respect to Normal, Upset, and Faulted Conditions. Stress limits for each of the loading combinations are component oriented and are presented in Table 3-90, Table 3-91, Table 3-92, Table 3-93, and Table 3-94 for tanks, inactive[1] pumps, active pumps, and valves, respectively. Active[2] pumps and valves are discussed in Section 3.9.3.2. Design of component supports is discussed in Section 3.9.3.4.

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[1] Inactive components are those whose operability are not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

[2] Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories and after a transient or event to place the plant in a safe shutdown condition. The safe shutdown condition is generally hot standby. In the case of a large LOCA the safe shutdown condition is the various recirculation modes.

The design stress limits established for the components are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are component oriented and are presented in Table 3-90 through Table 3-94.

3.9.3.1.2 Duke Class A Piping

The load combination requirements and corresponding stress criteria for Duke Class A (ASME Class 1) piping are presented in Table 3-95.

These loads are combined by direct summation of the pressure resultant loads and moments for sustained loads (M_A), occasional loads (M_B), and thermal expansion loads (M_C), as defined by their respective equations in subsection NB-3600 of ASME Section III, Division I.

A representative example for all Class A piping analysis performed by EDS Nuclear for Duke, was given to NRR as Appendix 3.9A to the FSAR prior to the License issuance. This example was a complete Class A stress report for a portion of the Boron Injection (NI) System, and included a summary of maximum stresses and cumulative usage factors for this portion of the (NI) system. The report also contained a description of the mathematical model, the method of analysis, a description of load conditions and the code compliance analysis results. This report can be found in the Duke Power Regulatory Compliance Licensing files.

3.9.3.1.3 Duke Class B, C, and F Piping

The load combination requirements and corresponding stress criteria for Duke Class B (ASME Class 2), Class C (ASME Class 3), and Class F (ANSI B31.1 seismic) piping are presented in Table 3-96.

These loads are combined by direct summation of the pressure loads and resultant moments for sustained loads (M_A), occasional loads (M_B), and thermal expansion loads (M_C) as defined by their respective equations in subsection NC or ND-3600 of ASME Section III, Division I or ANSI B31.1.

A representative example for all Class B, C, and F piping analysis performed by Duke is given in Figure 3-292 as an isometric of a portion of the Auxiliary Feedwater (CA) System pump discharge piping.

This typical piping analysis problem, with representative math model, was provided to NRR for illustration purposes prior to License issuance. The results of this analysis are available in Regulatory Compliance Licensing Files.

3.9.3.1.4 Design Loading Combinations and Design Stress Limits for Mechanical Equipment Furnished by Duke

The load combinations and corresponding stress criteria for Duke Mechanical equipment and valves are presented in Table 3-97.

3.9.3.1.5 Piping Supports and Restraints

The design loading combination associated with each component operating condition is given in Table 3-101 for supports, restraints, and anchors and in Table 3-102 for mechanical or hydraulic snubbers.

Loads for each loading combination are combined algebraically except that components which contain positive and negative values are combined to assemble the worst case load combination.

Design stress limits for each component operating condition are in accordance with Subsection NF of the ASME Boiler and Pressure Vessel Code for those portions of supports and restraints within the NF jurisdictional boundary. Stress limits for Normal and Upset Conditions are in accordance with Article XVII-2000. For Faulted Condition, design stress limits for manufacturer's standard support components are in accordance with the requirements of Appendix F. Emergency Condition stress limits, as specified in Article XVII-2000 are used for the design of all other components for Faulted Condition. Stresses for those portions of supports and restraints outside the NF jurisdictional boundary are limited to the allowable values in Table 3-101.

Snubbers are used at locations where restraints are necessary based on piping stress analysis, but thermal movement of the pipe must not be constrained. Performance selection is based on manufacturer's load capacity data and the requirement that the allowable travel of the snubber exceed the calculated pipe thermal travel. The midpoint of pipe thermal travel is set at the mid-point of the snubber travel range with hot and cold settings established accordingly. If snubbers are used to mitigate effects of operational vibration, the analytical and design methodology utilized as well as design specification requirements to assure that structural and mechanical performance characteristics and product quality are achieved will be developed and available for review.

Each snubber assembly is accessible after installation and all adjustment features are unobstructed and visible where possible. The manufacturer's figure number, size, stroke, and load rating is mounted on each snubber.

The loading combinations for Westinghouse items are given on Table 3-51.

3.9.3.2 Pump and Valve Operability Program

3.9.3.2.1 Westinghouse Pump and Valve Operability Program

Mechanical equipment classified as safety-related must be capable of performing its function under postulated plant conditions. Equipment with faulted condition function requirements include active pumps and valves in fluid systems important to safety. Seismic analysis is presented in Section 3.7 and covers all safety-related mechanical equipment. A list of all active pumps is presented in Table 3-103. Active valves are listed in Table 3-104.

All active pumps are qualified for operability by first being subjected to tests both prior to installation in the plant and after installation in the plant. The in-shop tests include 1) hydrostatic tests of pressure-retaining parts to 150 percent of the design pressure, 2) seal leakage tests at the same pressure used in the hydrostatic tests, 3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum heat, Net Positive Suction Head (NPSH) requirements and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer, based on the bearing material, clearances, oil type, and rotational speed.

After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and the required periodic in-service inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability during SSE condition by assuring that the pump will continue operating and not be damaged during the seismic event.

The pump manufacturer is required to show that the pump will operate normally when subjected to the maximum seismic accelerations and maximum faulted nozzle loads. It is required that tests or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 3g in the horizontal direction and 2g in the vertical direction acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited as indicated in Table 3-93. In addition, the pump casing stresses caused by the maximum seismic nozzle loads are limited to stresses outlined in Table 3-93. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3-93 as allowables, assures that critical parts of the pump will not be damaged during the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations are determined using the same conservatisms contained in the 3g horizontal and 2g vertical accelerations used for "rigid" structures. The static analysis is performed using the adjusted accelerations; the stress limits stated in Table 3-93 must still be satisfied.

The second criterion necessary to assure operability is that the pump will function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation. Typically, the rotor can be seized 5 full seconds before a circuit-breaker, to prevent damage to the motor, shuts down the pump. However, the high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from losing its function. In actuality, the seismic loadings will cause only a slight increase, if any, in the torque (i.e, motor current) necessary to drive the pump at the constant design speed. Therefore, the pump will not shutdown during the SSE and will operate at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is identified to be vital to the operation of the pump or pump motor, and which is not qualified for operation during the pump analysis or motor qualifications, is also separately qualified for operation at the accelerations it would seek at its mounting. The pump motor and vital auxiliary equipment is qualified by meeting the requirements of IEEE Standard 344-1971, with the additional requirements and justifications outlined in Section 3.9.3.2.2.

The program above gives the required assurance that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings, and, therefore, will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured since only normal operating loads and steady state

nozzle loads exist. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Active valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valve will open and close Qualification of motor operator for environmental conditions is discussed in Section 3.11. Cold hydro tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operations are performed in-situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant.

The valves are constructed in accordance with the ASME Boiler & Pressure Vessel Code, Section III. On active valves, an analysis of the extended structure is performed for static equivalent seismic loads applied at the center of gravity of the extended structure. The maximum stress limits used for active Class 2 and 3 valves are shown in Table 3-94.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated plant faulted condition by demonstrating operational capabilities within the specified limits. The testing procedures will be described below.

The valve is mounted in a manner which conservatively represents typical valve installations. The valve includes the operator normally attached to the valve in service. The faulted condition nozzle loads are limited to not effect the operability of the valve. The operability of the valve during a faulted condition is demonstrated by satisfying the following criteria:

1. Active valves are designed to have a first natural frequency which is greater than 33 Hz.
2. The actuator and yoke of the valve system is statically deflected an amount equal to the deflection caused by the faulted condition accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve will be applied to the valve during the static deflection tests.
3. The valve is cycled while in the deflected position. The time required to open or close the valve in the deflected position will be compared to similar data taken in the undeflected condition to evaluate the significance of any change.
4. Motor operators, pilot solenoid valves and external limit switches necessary for operation are qualified by IEEE standard 344-1971 with the additional requirements and justifications as supplied in Section 3.9.3.2.2.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to the simultaneous application of 3g in the horizontal direction and 2g in the vertical direction. The piping designer must maintain the accelerations to these levels.

The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types are tested. Valve sizes which cover the range of sizes in service are qualified by the tests and the results are used to qualify all valves within the intermediate range of sizes.

Using these methods, active valves are qualified for operability during a faulted event. These methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary.

Active pump motors (and vital pump appurtenances) and active valve operators, including limit switches and solenoid valves, are seismically qualified in accordance with IEEE standard 344-1971. If the testing option is chosen, sine-beat testing will be used. Seismic qualification by analysis alone or by a combination of analysis and testing, may be used when justified. The analysis program can be justified by: 1) demonstrating that equipment being qualified is amendable to analysis, and 2) that the analysis be correlated with test or be performed using standard analysis techniques.

3.9.3.2.2 Operability Assurance of Duke Safety-Related Active Pumps and Valves

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 times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, (b) seal leakage tests at the same pressure used in the hydrostatic 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.

   b. During and after faulted conditions, the safety related active pumps are qualified for operability by the tests and/or analyses described in Section 3.9.2.2 and the following:

      1) The pump manufacturer will be required to demonstrate that the lowest natural frequency of the pump is greater than 33 Hz. If all natural frequencies of the assembled equipment are greater than 33 Hz, the pump will be considered essentially rigid. If any natural frequencies of the assembled equipment are less than 33 Hz, the pump shall be analyzed dynamically or tested accordingly. Specific procedures for mathematical analysis are outlined in Section 3.9.2.2.1. Seismic qualification by testing, where practical, may be accomplished by following test procedures as outlined in Section 3.9.2.2.2.

      2) The stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the values given in Section 3.9.3.1. The maximum seismic nozzle loads are also considered in an analysis of pump supports to assure that a system misalignment cannot occur.

      3) The pump motor and all electrical appurtenances vital to operation of the pump are seismically qualified as described in Section 3.10. The shaft coupling between the pump and motor will be designed to minimize the coupling of the response of pump and motor.

      4) 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.

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 precluded 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 Catawba Nuclear Station safety-related pumps and system configurations in conjunction with manufacturer data/evaluations formed the basis of Duke Power Company'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.

2. Safety-Related Active Valves

The following criteria assure that safety-related active valves will function as designed:

a. The safety-related valves are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic tests to ASME Sections III requirements, backseat and main seat leakage tests, disc hydrostatic test, functional tests to verify that the valve will open and close within the specified time limits when subjected to the design differential pressure, operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, etc.) according to IEEE 382-1972. Cold hydro qualification tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation are performed in-situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are designed using either stress analyses or the pressure-containing minimum wall thickness requirements.

The NRC issued Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing," on June 28, 1989, to require in-situ testing and verification of design basis capability of motor-operated valves to perform their required safety functions. Applicable testing and design basis verifications efforts proceeded through several Unit 1 and Unit 2 outages, with final response to the NRC provided in the letter from W.R. McCollum, Jr. to the NRC, dated February 20, 1997. This program followed on and encompassed earlier analyses, inspections and testing to address NRC valve limit switch setting issues raised in IE Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings." 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 30. The NRC accepted Catawba's Generic Letter 96-05 program and documented closure of the Generic Letter 89-10 program in Reference 31."

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 that licensees take actions to ensure that safety-related power-operated gate valve that are susceptible to pressure locking or thermal binding are capable of performing their safety functions. Evaluations of the population of valves within this category were completed

with responses to the NRC submitted in the letter from M.S. Tuckman to the NRC, dated February 13, 1996, and the letter from W.R. McCollum to the NRC, dated July 31, 1996. No valves were found to be susceptible to thermal binding or to thermally-induced pressure locking. Additionally, no valves were found to be incapable of performing their safety open functions in the event they saw hydraulically-induced pressure locking. These responses also included commitments to test/modify several valves during future outages in order to enhance their performance.

b. During and after faulted conditions, the safety-related active valves are qualified for operability by the tests and/or analyses described in Section 3.9.2.2 and the following:

1) The valve manufacturer will be required to demonstrate that the natural frequency of the valve is greater than 33 Hz. If all natural frequencies of the assembled equipment are greater than 33 Hz, the valve will be considered essentially rigid. If any natural frequencies of the assembled equipment are less than 33 Hz, the valve shall be analyzed dynamically or tested accordingly.

   Specific procedures for mathematical analysis are outlined in Section 3.9.2.2.1. Seismic qualification by testing, where practical, may be accomplished by following test procedures as outlined in Section 3.9.2.2.2.

2) During testing the valve will be mounted in a manner which will be conservatively representative of a typical valve installation.

3) While installed in a suitable test rig, the extended top works (i.e. the actuator and the yoke) of the valve are statically loaded with an amount greater than the equivalent seismic loads (SSE) as determined by analysis described in Section 3.9.2.2. The load is applied at the center of gravity of the operator in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static load tests. With the equivalent seismic static load applied, the valve is then operated from normal operating mode to the faulted operating mode. The valve must perform its safety-related functions within the specified operating time limits.

4) The accelerations used in the qualification test of step c will be provided to the piping designer who must maintain the valve operator accelerations to these levels with adequate margin of safety.

5) Motor operators and other electrical appurtenances necessary for operation are seismically qualified as described in Section 3.11.

6) Steps c, d and e above apply only to valves with overhanging structures (i.e., the motor operator). The testing will be conducted on a representative number of valves. Valves from each of the primary safety-related design types (e.g., motor-operated gate valve) will be tested. Valve sizes which cover the range of sizes in service will be qualified by the tests and the results will be used to qualify all valves within the intermediate range of sizes. Stress analyses will be used to support the interpolation.

7) Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves, will be considered separately. Due to the particularly simple characteristics of check valves and other compact valves, they are qualified by a combination of the following tests and analyses:

   a) Stress analysis including seismic analysis as described in Section 3.9.2.2.1.

   b) In-shop hydrostatic test

c) In-shop leakage test

d) Periodic valve exercise and inspection to assure functional ability.

3.9.3.2.3 Functional Capability

On Catawba Nuclear Station Duke will apply the following criteria to essential piping in order to assure that functional capability is maintained when subjected to loads in excess to those for which Upset Limits are specified.

1. Reference 15 will be applied to determine the acceptance of functional capability for ASME Class 1 piping and fittings.
2. ASME Class 2 and 3 piping and fittings with the exception of stainless steel elbows will be accepted as meeting functional capability when the following equation is met:

$$\frac{P \max Do}{4t_n} + \frac{0.75i(MA + 1.875MB)}{Z} \leq 1.8Sh \qquad \text{[Equation 2]}$$

3. ASME Class 2 and 3 piping and fittings that do not meet equation 1 will be accepted as meeting functional capability when $Do / tn \leq 50$ and the following equation is met.

$$\frac{B1P \max Do}{2t_n} + \frac{B2(MA + 1.875MB)}{Z} \leq 2.25Sh \text{ But not greater than } 1.8\,Sy \qquad \text{[Equation 2]}$$

Eq. 9 ASME Code NC-3600 W'81 addenda

4. ASME Class 2 and 3 stainless steel elbows will be accepted as meeting functional capability when Do/tn < 50 and the following equation is met.

$$\frac{B1P \max Do}{2t_n} + \frac{B2(MA + 1.875MB)}{Z} \leq 1.8Sy \qquad \text{[Equation 3]}$$

Eq. as developed by Westinghouse for Comanche Peak Steam Electric Station.

5. Reference 15 will be applied to determine the acceptance of functional capability for ASME Class 2 and 3 stainless steel elbows and those that do not meet equation 1 when 50 < Do/tn < 100.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

3.9.3.3.1 Main Steam System

Overpressure protection for the Main Steam System is provided by the main steam safety valves. As further described in Chapter 10, sufficient relief capacity is provided to meet ASME overpressure protection requirements. The safety valves are set for progressive relief at increasing pressures within the code - allowed range, to avoid valve chattering and to minimize the chance that more than one valve will actuate simultaneously. The valves are located in the main steam dog-house structure, and discharge to open stacks. The geometry of the valve discharge piping is such that the steady state thrust does not produce bending in the valve inlet nozzle. These nozzles are reinforced in accordance with standard Code practice for internal pressure.

The structural integrity of the main steam safety valves and their mountings is assured using a conservative analysis. Methods are in accordance with Regulatory Guide 1.67. Credit is not taken for some conservative design features such as staggered set pressure. All valves on a given header are assumed to initiate discharge instantaneously (dynamic load) and simultaneously.

The steady state thrust is calculated based on sonic velocity at the valve orifice and at the discharge pipe outlet. This model employs the principles of continuity of flow, isentropic flow from pressure source to valve, and isenthaplic throttling across the valve. Thrust results are compared with data supplied by the safety valve manufacturer.

In other respects, the analysis of these valves and their piping is the same as for any safety-related system. Valve thrust is recognized as one of the load components of the Upset and Faulted plant conditions. The criteria for load combinations and allowable stresses are discussed in Section 3.9.3.1. The results of the main steam safety valve analyses must be that the valves, piping and supports are within applicable stress limits for all the postulated conditions.

3.9.3.3.2 Pressurizer Safety and Relief Devices

The Pressurizer Relief System, is a closed system which consists of three power operated relief valves with opening times of 3 seconds, three safety valves which have set pressures and interconnective piping which extends from the pressurizer to the pressurizer relief tanks. These valves and Relief Piping System are designed to meet the ASME Code requirements for over-pressure protection. The three power operated relief valves are actuated by control room signals while the safety valves will open independently when their set pressures are exceeded.

To assure an accurate assessment of the Pressurizer Relief System structural response during and following the actuation of the safety and relief valves, detailed thermal-hydraulic and structural dynamic time-history analyses are performed. To obtain maximum loadings during the valve discharge transients, it is assumed that all safety and relief valves commence opening simultaneously. The thermal hydraulic dynamic analyses are performed using the RELAP computer code.

The flow induced force time-histories are then used in a dynamic time-history structural analysis of the entire Pressurizer Relief System using the computer program SUPERPIPE. The structural analysis results in the determination of time-histories of displacements, stresses and support reaction forces throughout the Pressurizer Relief System. For conservatism in combination with other loading conditions, the maximum stresses and reaction forces determined from this analysis are combined without regard to sign or differences in time of occurrence.

Normal operating conditions for the Pressurizer Relief System consist of internal pressure, dead weight, transient and steady state thermal loads and the system transient response to valve operation.

To assure compliance with the stress limits of the ASME Code for the Class 1 and 2 components of the Pressurizer Relief System, the following operating conditions, in addition to the normal operating conditions noted above are evaluated:

Upset Condition: Normal Condition Loads and Operating Basis Earthquake (OBE).

Faulted Conditions: Normal Condition Loads and Safe Shutdown Earthquake (SSE).

Stress computations and stress limit evaluations are performed in accordance with the ASME Code requirements. Design and analysis iterations of the Pressurizer Relief System are conducted, as necessary, to ensure compliance with the ASME Code limits.

3.9.3.3.3 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-I-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-I-89-25 and III-I-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 10CFR50.55a(d) and (e) apply to power plants whose applications for construction permits were docketed after May 14, 1984. Therefore, the regulations in 10CFR50.55a(d) and (e) concerning ASME Section III design requirements do not apply to McGuire and Catawba. Nevertheless, ASME Code systems' design at Catawba is a commitment per UFSAR Table 3-5. This NRC letter advised that Duke may treat this non-compliance with ASME Code commitments in UFSAR 3.0 as a "change to the facility as described in the SAR", and accept this non-compliance as acceptable under 10CFR50.59. Consequently, 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 design basis documents and procedures) and physical barriers such as locks.

3.9.3.4 Component Supports

Loading combinations, design transients, and stress limits for balance of plant component supports are discussed in Section 3.9.3.1. The use of these criteria provide a conservative basis for assuring no loss of structural integrity to supports and restraints, even under adverse loading conditions.

For Westinghouse supplied Class 2 and 3 component supports loading combinations are defined in Table 3-88. The stress limits applicable for Class 2 and 3 component supports are as follows:

1. Linear Supports for Tanks and Heat Exchangers
    a. Normal - The allowable stresses of A.I.S.C.-69 Part 1 are employed for normal condition allowables.
    b. Upset - Stress limits for upset conditions are 33 percent higher than those specified for normal conditions. This is consistent with Paragraph 1.5.6 of A.I.S.C.-69 Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

   c. Faulted - Stress limits for faulted condition are the same as for the upset condition.

2. Plate and Shell Supports for Tanks and Heat Exchangers

   a. Normal - Normal condition limits are those specified in ASME Section VIII, Division 1 or A.I.S.C.-69 Part 1.

   b. Upset - Stress limits for upset condition are 33 percent higher than those specified for normal conditions. This is consistent with Paragraph 1.5.6 of A.I.S.C.-69 Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

   c. Faulted - Stress limits for faulted condition are the same as for the upset condition.

3. Plate and Shell Supports for Pumps - The stress limits used for ASME Code Class 2 and 3 plate and shell component supports are identical to these used for the supported component. These allowable stresses are such that the design requirements for the components and system assume that structural integrity is maintained.

3.9.4 Control Rod Drive System (CRDS)

3.9.4.1 Descriptive Information of CRDS

Control Rod Drive Mechanism

Control rod drive mechanisms are located on the dome of the reactor vessel.

They are coupled to rod control clusters which have absorber material over the entire length of the control rods. The control rod drive mechanism is shown in Figure 3-293 and schematically in Figure 3-294.

The primary function of the control rod drive mechanism is to insert or withdraw rod cluster control assemblies within the core to control average core temperature and to shutdown the reactor.

The control rod drive mechanism is a magnetically operated jack. A magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod cluster control assemblies in the reactor core in discrete steps. Rapid insertion of the rod cluster control assemblies occurs when electrical power is interrupted.

The control rod drive mechanism consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

1. The pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal welded, maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded plug with a canopy seal weld for pressure integrity. This closure contains a threaded plug used for venting.

   The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

2. The coil stack assembly includes the coil housings, and electrical conduit and connector, and three operating coils: 1) the stationary gripper coil, 2) the movable gripper coil, and 3) the lift coil.

   The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: 1) the movable gripper latches and 2) the stationary gripper latches.

   The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8 inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8 inch step.

4. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The driver rod has 5/8 inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and provides the means for coupling to the rod cluster control assembly.

   The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod cluster control assembly and permits remote disconnection of the drive rod.

   The control rod drive mechanism is a trip design. Tripping can occur during any part of the power cycle sequencing if electrical power to the coils is interrupted.

   The control rod drive mechanism is threaded and seal welded on an adaptor on top of the reactor vessel and is coupled to the rod cluster control assembly directly below.

   The mechanism is capable of raising or lowering a 360 pound load, (which includes the drive rod weight) at a rate of 45 inches/minute. Withdrawal of the rod cluster control assembly is accomplished by magnetic forces while insertion is by gravity.

   The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F with forced air cooling required to maintain the coils below or at 392°F.

   The control rod drive mechanism shown schematically in Figure 3-294 withdraws and inserts a rod cluster control assembly as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

   During plant operation the stationary gripper coil of the drive mechanism holds the rod cluster control assembly in a static position until a stepping sequence is initiated at which time the movable gripper coil and lift coil is energized sequentially.

<u>Rod Cluster Control Assembly Withdrawal</u>

The rod cluster control assembly is withdrawn by repetition of the following sequence of events (refer to Figure 3-294).

1. Movable Gripper Coil (B) - ON

The latch locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 1/16 inch axial clearance exists between the latch teeth and the drive rod.

2. Stationary Gripper Coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

3. Lift Coil (C) - ON

   The 5/8 inch gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 inch).

4. Stationary Gripper Coil (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing and the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

5. Movable Gripper Coil (B) - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

6. Lift Coil (C) - OFF

   The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

7. Repeat Step 1

   The sequence described above (Items 1 through 6) is termed as one step or one cycle. The rod cluster control assembly moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5/8 inch groove pitch) is raised 72 grooves per minute. The rod cluster control assembly is thus withdrawn at a rate up to 45 inches per minute.

<u>Rod Cluster Control Assembly Insertion</u>

The sequence for rod cluster control assembly insertion is similar to that for control rod withdrawal, except the timing of lift coil (C) ON and OFF is changed to permit lowering the control assembly.

1. Lift Coil (C) - ON

   The 5/8 inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Movable Gripper Coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 1/16 inch axial clearance exists between the latch teeth and the drive rod assembly.

3. Stationary Gripper Coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached rod cluster control assembly, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached rod cluster control assembly is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift Coil (C) - OFF

   The force of gravity and spring force separates the movable gripper pole from the lift pole and the drive rod assembly and attached rod cluster control drop down 5/8 inch.

5. Stationary Gripper (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable Gripper Coil (B) - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Repeat Step 1

   The sequence is repeated, as for rod cluster control assembly withdrawal, up to 72 times per minute which gives an insertion rate of 45 inches per minute.

Holding and Tripping of the Control Rods

During most of the plant operating time, the control rod drive mechanisms hold the rod cluster control assemblies withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached rod cluster control assemblies hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the rod cluster control assembly plus the stationary gripper return spring is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight stationary gripper return spring and weight acting upon the latches. After the rod cluster control assembly is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9.4.2 Applicable CRDS Design Specifications

For those components in the Control Rod Drive System comprising portions of the reactor coolant pressure boundary, conformance with the General Design Criteria and 10CFR 50, Section 50.55a is discussed in Sections 3.1 and 5.2 conformance with Regulatory Guides pertaining in Sections 4.5 and 5.2.3.

Design Bases:

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

Design Stresses:

The Control Rod Drive System is designed to withstand stresses originating from various operating conditions as summarized in Table 3-88. The CRDS has been conservatively designed to withstand the stresses originating from a LOCA (full double-ended RCS primary loop pipe break) even though such pipe breaks are no longer considered for dynamic effects according to Reference 16.

Allowable Stresses:

For normal operating conditions Section III of the ASME Boiler and Pressure Code is used. All pressure boundary components are analyzed as Class I components.

Dynamic Analysis: The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the Control Rod Drive System.

Control Rod Drive Mechanisms

The control rod drive mechanisms (CRDM's) pressure housings are Class I components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

The control rod drive mechanisms (CRDMs) are evaluated for the effects of postulated reactor vessel inlet nozzle and outlet nozzle limited displacement breaks. A time history analysis of the CRDMs is performed for the vessel motion discussed in Section 3.9.1.4.5. A model of the CRDMs is formulated with gaps at the upper CRDM support modeled as nonlinear elements. The CRDMs are represented by beam elements with lumped masses. The translation and rotation of the vessel head is applied to this model. The resulting loads and stresses are compared to allowables to verify the adequacy of the system.

A dynamic seismic analysis is required on the CRDM's when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

Full Length Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the full length CRDM's are:

1. 5/8 inch step,
2. Approximately 144 inch,
3. 360 pound maximum load,
4. Step in or out at 45 inches/minute (72 steps/minute),
5. Electrical power interruption shall initiate release of drive rod assembly,
6. Trip delay time of less than 150 milliseconds - Free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F,
7. 40 year design life with normal refurbishment.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9.4.3.1 Pressure Vessel

The pressure retaining components 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 analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control Rod Trip (equivalent static load)
2. Differential Pressure
3. Spring Preloads
4. Coolant Flow Forces (static)
5. Temperature Gradients
6. Differences in thermal expansion
    a. Due to temperature differences
    b. Due to expansion of different materials
7. Interference between components
8. Vibration (mechanically or hydraulically induced)
9. Operational transients as indicated in Table 3-50
10. Pump Overspeed
11. Seismic Loads (operation basis earthquake and design basis earthquake)
12. Blowdown Forces (due to cold and hot leg break)

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components.

3.9.4.3.2 Drive Rod Assembly

All postulated failures of the drive rod assemblies either by fracture or uncoupling lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with, and is guided by the rod cluster control assembly. This always results in reactivity decrease.

3.9.4.3.3 Latch Assembly and Coil Stack Assembly

<u>Results of Dimensional and Tolerance Analysis</u>

With respect to the control rod drive mechanism system as a whole, critical clearances are present in the following areas:

1. Latch assembly (Diametral clearances)
2. Latch arm-drive rod clearances
3. Coil stack assembly-thermal clearances
4. Coil fit in coil housing

The following write-up defines clearances that are designed to provide reliable operation in the control rod drive mechanism in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 68°F is 0.011 inches. At the maximum design temperature of 650°F minimum clearance is 0.0045 inches and at the maximum expected operating temperatures of 550°F is 0.0057 inches.

Latch Arm - Drive Rod Clearances

The control rod drive mechanism incorporates a load transfer action. The movable or stationary gripper latch are not under load during engagement, as previously explained, due to load transfer action. Figure 3-295 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3-296 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F the inside diameter of the coil stack is 7.308/7.298 inches. The outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism due to operating temperature of the control rod drive mechanism result in minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed from four hot control rod drive mechanisms mounted on 11.035 inch centers on a 550° test loop, allowed to cool, and then placed without incident as a test to prove the preceding.

Coil Fit in Core Housing

Control rod drive mechanism and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot control rod drive mechanism.

3.9.4.3.4 Evaluation of Control Rod Drive Mechanisms and Supports

The control rod drive mechanisms (CRDMs) and CRDM support structures are evaluated for the loading combinations outlined in Table 3-51.

A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis,

nonlinearities in the structure are represented. These include RPI plate impact, tie rods, and lifting leg clevis/RPV head interface. The time history motion of the reactor vessel head, obtained from the RPV analysis described in Section 3.9.1.4.5, is input to the dynamic model. Maximum forces and moments in the CRDMs and support structure are then determined. For the seismic analysis, the structural model is linearized and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDMs for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evaluation of the CRDMs. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

The highest loads occur at the head adaptor, the location where the mechanisms penetrate the vessel head. The bending moments at this location are presented in Table 3-83 for the longest and shortest CRDM.

3.9.4.4 CRDS Performance Assurance Program

Evaluation of Material's Adequacy

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Boiler and Pressure Vessel Code, Section III.

Internal components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment as confirmed by life tests (Reference 11). Latch assembly inspection is recommended after 2.5 E6 steps have been accumulated on a single control rod drive mechanism.

HISTORICAL INFORMATION NOT REQUIRED TO BE UPDATED.

To confirm the mechanical adequacy of the fuel assembly, the control rod drive mechanism, and rod cluster control assembly, functional test programs have been conducted on a full scale 12 foot control rod. The 12 foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the full length control rod drive mechanisms meet the design requirement of 2.2 seconds from start of decay of stationary gripper coil voltage to dashpot entry, at full flow and operating temperature. This trip time requirement will be confirmed for each control rod drive mechanism prior to initial reactor operation and at periodic intervals after initial reactor operation as required by the Technical Specifications.

There are no significant differences between the prototype control rod drive mechanisms and the production units. Design materials, critical tolerances and fabrication techniques are the same.

The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors. These tests have been reported in Reference 11.

In addition, dynamic testing programs have been conducted by Westinghouse and Westinghouse Licensees to demonstrate that control rod scram time is not adversely affected by postulated seismic events. Acceptable scram performance is assured by also including the effects of the allowable displacements of the driveline components in the evaluation of the test results.

It is expected that all control rod drive mechanisms will meet specified operating requirements for the duration of plant life with normal refurbishment. However, a technical specification pertaining to an inoperable rod cluster control assembly has been set. Latch assembly inspection is recommended after 2.5 E6 steps have been accumulated on a single control rod drive mechanism.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited to one as discussed in the Technical Specifications.

In order to demonstrate proper operation of the Control Rod Drive Mechanism and to ensure acceptable core power distributions during rod cluster control assembly partial-movement checks are performed on the rod cluster control assemblies. (Refer to Technical Specifications.) In addition, periodic tests of the rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical rod cluster control assembly ejection. During these tests the acceptable drop time of each assembly is not greater than 2.2 seconds, at full flow and operating temperature, from the beginning of decay of stationary gripper coil voltage to dashpot entry.

Actual experience in operating Westinghouse plants indicates excellent performance of control rod drive mechanisms.

All units are production tested prior to shipment to confirm ability of the control rod drive mechanism to meet design specification-operation requirements.

Each production control rod drive mechanism undergoes a production test as listed below:

Test	Acceptance Criteria
Cold (ambient) hydrostatic	ASME Section III
Confirm step length and load transfer (stationary gripper to movable gripper or movable gripper to stationary gripper)	Step Length 5/8 ± 0.015 inches axial movement Load Transfer 0.047 inches nominal axial movement
Cold (ambient) performance Test at design load - 5 full travel excursions	Operating Speed 45 inches/minute Trip Delay Free fall of drive rod to begin within 150 MS

The NRC issued Generic Letter 93-04, "Control Rod System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)," on June 21, 1993. This generic letter notified licensees of the occurrence at the Salem Nuclear Generating Station, Unit 2, of a control rod position indicator malfunction during reactor startup which had the possibility of placing that unit outside its fuel design bases (rod withdrawal actual exceeding indicated). Accordingly, the NRC requested review of the similar Catawba Nuclear Station systems supplied by Westinghouse. As stated in the initial responses to Generic Letter 93-04 (letter from M.S. Tuckman to the NRC, dated August 5, 1993; and letter from M.S. Tuckman to the NRC, dated September 20, 1993), the evaluations presented for similar control rod equipment failures concluded that Departure from Nucleate Boiling (DNB) would not occur for the worst-case asymmetric rod withdrawal. Short-term compensatory measures were identified which were implemented by CNs to preclude an event similar to that described in the generic letter. Further long-term actions committed to were to implement the Westinghouse Owner's Group (WOG) recommended current order timing modification and the new current order surveillance test. These long-term actions were verified completed for Units 1 and 2 in the letter from W.R. McCollum to the NRC, dated December 1995.

3.9.5 Reactor Vessel Internals

3.9.5.1 Design Arrangements

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel

to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3-297. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guides by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in Reference 12.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and then through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by “key” and “keyway” joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Another Inconel insert block is bolted to each of these blocks and has a “keyway” geometry. Opposite each of these is a “key” which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal

structures are within the ASME Section III limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy absorbing devices of the internals to the vessel.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. The main purpose of the energy absorber is to absorb impact loads of the core and the supporting structure during a postulated core drop accident. The energy of the impact is absorbed by an energy absorbing mechanism which consists of a "necked-down" portion as shown in Figure 3-297. Using energy principles the total potential energy of the system is absorbed by the strain energy of the energy-absorbing devices.

The maximum deformations of the energy absorbing assemblies during the core drop accident remain well within the functional limits so as not to affect the RCS Scram Function. It should also be noted that the maximum strains undergone by the energy absorbers (≤ 15% strains, hot condition) during the core drop accident as well below the fracture limits (62% strains for 304 stainless steel at 600°F) and satisfy the SRP requirements.

Upper Core Support Assembly

The upper core support assembly, shown in Figure 3-298 and Figure 3-299 consists of the top support plate assembly, and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate assembly and the upper core plate and are fastened at top and bottom to these plates. The UHI support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. They position the upper core plate and upper support which act as the boundaries for the flow plenum at the outlet of the core. Additionally each UHI column has a central axial flow passage full length for conveying core cooling water to the core when it is injected into the vessel head. The water enters the flow passage through a small hole on the side of the top of the UHI support column. A support column is provided at each fuel assembly position that does not contain accommodation for a control rod with the exception of the peripheral low power fuel assembly locations. The fuel assemblies which do not have a support column above them are located in front of the inlet and outlet nozzles of the vessel. The UHI support columns also contain thermocouple supports.

The guide tube assemblies shield and guide the control rod drive shafts and control rods. They are fastened to the top support plate and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube. In the UHI system, the guide tubes also serve to transport UHI water from the vessel head region to the area directly above the fuel assemblies. All units having UHI have the maximum number of guide tubes independent of other RCC requirements.

The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90° from each other. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods are

thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the top support plate assembly and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7-26 shows the Basic Flux-Mapping System).

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes. In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the training ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the incore instrumentation support structure.

3.9.5.2 Design Loading Conditions

The design loading conditions that provide the basis for the design of the reactor internals are:

1. Fuel Assembly Weight
2. Fuel Assembly Spring Forces
3. Internals Weight
4. Control Rod Trip (equivalent static load)
5. Differential Pressure

6. Spring Preloads
7. Coolant Flow Forces (static)
8. Temperature Gradients
9. Differences in thermal expansion
   a. Due to temperature differences
   b. Due to expansion of different materials
10. Interference between components
11. Vibration (mechanically or hydraulically induced)
12. One or more loops out of service
13. Operational transients as indicated in Table 3-50
14. Pump overspeed
15. Seismic loads (operation basis earthquake and safe shutdown earthquake)
16. Blowdown forces - injection transients for the cold and hot leg break including UHI

The main objective of the design analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analysis on the reactor internals are provided in Section 3.9.2.

As part of the evaluation of design loading conditions, extensive testing and inspections are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

3.9.5.3 Design Loading Categories

The combination of design loadings fit into either the normal, upset, or faulted conditions as defined in the ASME Code, Section III.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3-50.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss of coolant accident plus the safe shutdown earthquake condition, the deflection criteria of critical internal structures are limiting values given in Table 3-87. The corresponding no loss of function limits are included in Table 3-87 for comparison purposes with the allowed criteria. The reactor internals structures have been conservatively designed to withstand the stresses originating from a LOCA (full double-ended RCS primary loop pipe break) even though such pipe breaks are no longer considered for dynamic effects, according to Reference 16.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately one half inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inches which is insufficient to permit the trips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are 4 supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive stop is provided to ensure support.

3.9.5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internal components are as follows:

1. The reactor internals in conjunction with the fuel assemblies shall direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.
2. In addition to neutron shielding provided by the reactor coolants, a separate neutron pad assembly is provided to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.
3. Provisions shall be made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.
4. The core internals are designed to withstand mechanical loads arising from operating basis earthquake, design basis earthquake and pipe ruptures and meet the requirement of Item 5 below.
5. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.
6. Following the design basis accident, the plant shall be capable of being shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits.

This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

7. UHI upper internals assembly are to provide passage for the core cooling water from the vessel head plenum directly to the top of the fuel assemblies during the postulated loss of coolant accident.

The functional limitations for the core structures during the design basis accident are shown in Table 3-87. To ensure no column loading or rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 3-87.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9.2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions Section III of the ASME Nuclear Power Plant Components Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered.

It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design basis accident used for the core supports structures are based on the January, 1971 draft of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

The reactor internals for Catawba were fabricated prior to Subsection NG of the ASME Code becoming a requirement. However, with the exception of the Code Stress Report and Code Stamp, the reactor internals satisfy the design and fabrication requirements of Subsection NG of the ASME Code.

3.9.6 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 testing of these components is required by 10 CFR 50.55 a (f). In accordance with paragraph (f) (3) 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 Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular pump or valve. At Duke Power Company option, a later edition of the code may be adopted.

Throughout the service life of the station, the inservice testing program shall be updated to the requirements, except design and access provisions and preservice examination requirements, set forth in editions and addenda of the ASME Operation and Maintenance (OM) Code in accordance with 10CFR 50.55a(f)(4).

3.9.6.1 Inservice Testing of Pumps

Inservice testing of all safety-related ASME Class 1, 2 or 3 pumps that are provided with an emergency power source will be conducted as required by 10 CFR 50.55a. Preoperational

tests will establish reference values of such parameters as speed, pressure, flow rate, vibration, lubrication, and bearing temperature at normal operating conditions. Specific details which identify each pump to be tested, test parameters to be measured, test frequency and the edition of the ASME Code to be utilized for the initial inservice examinations will be submitted for NRC Review at least six months prior to the anticipated receipt of an operating license. If conformance with certain code requirements are determined to be impractical, information supporting these determinations will also be submitted at that time.

3.9.6.2 Inservice Testing of Valves

Inservice testing of safety-related ASME Class 1, 2 or 3 valves will be conducted as required by 10 CFR 50.55a. Valves which are used for operating convenience only such as manual vent, drain, instrument and test or valves utilized for maintenance only will be excluded from this program. Preoperational tests will establish reference valves such as valve stroke time. Specific details which identify valves to be tested, type of test to be performed, frequency of test and the edition of the ASME Code to be utilized for the initial inservice examinations will be submitted for NRC review at least six months prior to the anticipated receipt of an operating license. If conformance with certain code requirements are determined to be impractical, information supporting these determinations will also be submitted at that time.

The NRC issued Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," on April 3, 1989. It was determined in subsequent correspondence between the NRC and Duke Power Company that no Confirmatory Action Response was required for Catawba Nuclear Station. However, the following Inservice Testing (IST) Program change commitments were made in the letter from H.B. Tucker to the NRC, dated February 14, 1990. These items are extracted from Attachment 1 of Generic Letter 89-04:

1. Item 1 (Full Flow Testing of Check Valves) and Item 2 (Alternate to Full Flow Testing of Check Valves) to be incorporated into Catawba's IST program by the end of the Unit 2 EOC3 outage.
2. Item 3 (Back Flow Testing of Check Valves) to be incorporated into Catawba's IST program. A review of all check valves to determine the applicability of backflow testing will be completed by January 1, 1991. Procedures will be developed and implemented by the applicable U1EOC5 and U2EOC4 outages.
3. Item 5 (Limiting Values of Full-Stroke Times for Power Operated Valves) to be incorporated into Catawba's IST program by January 1, 1991.
4. Review of further practical enhancements in line with Generic Letter 89-04 as part of the end of first 10-year IST review process.

3.9.6.3 Relief Requests

As noted in Section 3.9.6, the inservice testing program will be periodically updated to meet requirements of the ASME Code. However, if it proves impractical to implement those criteria; requests for relief will be submitted on a case-by-case basis.

3.9.7 References

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2. "Benchmark Problem Solutions Employed for Verification of WECAN Computer Program," WCAP-8929, April, 1977.

3. "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, AMSE Publication, 1972.

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5. *Bogard, W.T., Esselman, T.C., "Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants," WCAP-9279, March 1978.*

6. *Bloyd, C.N., Ciarametars, W., and Singleton, N.R., "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8780, May 1976.*

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9. Bohm, G.J. and LaFaille, J.P., "Reactor Internals Response Under a Blowdown Accident," First Intl. Conf. on Structural Mechanics in Reactor Technology, Berlin, September 20-24, 1971.

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14. Cloud, R. L., "Damping Valves of Nuclear Plant Components," WCAP-7921-AR May, 1974.

15. NEDO-21985 "Functional Capability Criteria for Essential Mark II Piping," General Electric Company San Jose, CA 95125.

16. Letter from H.B. Tucker (DPC) to E.G. Adensam (NRC), dated May 11, 1984, transmitting Westinghouse report justifying elimination of RCS primary loop breaks for certain design considerations.

17. BWNT Computer Software Manual for Program NPD-TM-35, "BWSPAN, Linear Static and Dynamic Analysis Program User's Manual", Manual Revision I, Software Version 3.2HP (April 1993) and Manual Revision J, Software Version 4.0HP (August 1993).

18. BWMT 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).

19. Nuclear Regulatory Commission, Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, from David B. Matthews, May 5, 1988, NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss."
20. Duke Power Company, Letter from H.B. Tucker to NRC, January 15, 1990, re: Response to NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss," Action 4 Final Report.
21. Duke Power Company, Letter from M.S. Tuckman to NRC, January 10, 1991, re: Response to NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss," Description of Actions Completed or in Progress.
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26. Duke Power Company, Letter from M.S. Tuckman to NRC, February 13, 1996, re: Response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves."
27. Duke Power Company, Letter from W.R. McCollum to NRC, July 31, 1996, re: Response to Request for Additional Information - Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves," Catawba Nuclear Station, Units 1 and 2.
28. Nuclear Regulatory Commission, Letter to All Holders of Light Water Reactor Operating Licenses and Construction Permits, from Steven A. Varga, April 3, 1989, "Guidance on Developing Acceptable Inservice Testing Programs (Generic Letter No. 89-04)."
29. Duke Power Company, Letter from H.B. Tucker to the NRC, February 14, 1990, re: NRC Generic Letter 89-04 and NRC Inspection Reports Nos. 50-413, 414/89-27.
30. M.S. Tuckman (Duke) letter to Document Control Desk (NRC) dated March 17, 1997, *180-day Response to Generic Letter 96-05: Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves*, Docket Nos. 50-269, -270, -287, -369, -370, -413, -414.
31. P. Tam (NRC) letter to G.R. Peterson (Duke) dated May 14, 1999, *Catawba Nuclear Station Units 1 and 2, RE: Safety Evaluation on Your Response to Generic Letter 96-05*, Docket Nos. 50-413, -414.
32. Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Request for Relief Number 06-CN-003, Use of Polyethylene Materials in Nuclear Safety Related Piping Applications, Dated October 26, 2006.

33. Catawba Nuclear Station, Units 1 and 2, Relief 06-CN-003 Use of Polyethylene Material in Nuclear Service Water Applications (TAC Nos. MD3729 and MD 3730), Dated September 12, 2008.

34. Catawba Nuclear Station, Units 1 and 2, Relief 06-CN-003 For Use of Polyethylene Material in Buried Service Water Piping (TAC Nos. ME0234 and ME0235), Dated May 27, 2009.

THIS IS THE LAST PAGE OF THE TEXT SECTION 3.9.

3.10 Seismic Qualification of Seismic Category 1 Instrumentation and Electrical Equipment

Note:

This section of the FSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document (DBD) for this system/structure.

This section presents information to demonstrate that instrumentation and electrical equipment classified as seismic Category I is capable of performing designed safety-related functions in the event of an earthquake.

Seismic Category I instrumentation and electrical equipment include the following:

1. Pressure, Temperature, and Flow Transmitters (Nuclear Safety Related)
2. Resistance Temperature Detectors (Nuclear Safety Related)
3. Temperature and Pressure Controllers (Nuclear Safety Related)
4. Temperature, Pressure, and Level Switches (Nuclear Safety Related)
5. PAM Recorders and Indicators
6. Excore Neutron Power Range Detectors (Nuclear Safety Related)
7. Process Protection System Cabinets
8. Nuclear Instrumentation System Cabinets
9. Solid State Protection System Cabinets
10. Reactor Trip Switchgear
11. Engineered Safeguards Test Cabinets
12. Pump, Fan, and HVAC Motors (Nuclear Safety Related)
13. Valve/Damper Motor and Solenoid Operators (Nuclear Safety Related)
14. Diesel Generator and Accessories
15. Diesel Generator Support Equipment (Nuclear Safety Related)
16. 4160 Vand 600 V Switchgear (Nuclear Safety Related)
17. 600 V Motor Control Centers (Nuclear Safety Related)
18. 4160/600 V Transformers (Nuclear Safety Related)
19. 120 VAC Transformers (Nuclear Safety Related)
20. 125 VDC Batteries, Battery Chargers, and Inverters (Nuclear Safety Related)
21. 125 VDC Distribution Centers (Nuclear Safety Related)
22. 125 VDC and 120 VAC Panelboards (Nuclear Safety Related)
23. Containment Electric Penetrations (Nuclear Safety Related)
24. Main Control Boards
25. Auxiliary Shutdown Panel (Nuclear Safety Related)

26. Auxiliary Feedwater Turbine Panel
27. Miscellaneous Control Boards and Electrical Enclosures (Nuclear Safety Related)
28. Radiation Monitors (Nuclear Safety Related)
29. Electric Hydrogen Recombiners
30. Cable Tray and Supports (Nuclear Safety Related)

3.10.1 Seismic Qualification Criteria

3.10.1.1 Qualification Standards

The methods of meeting the general requirements for seismic qualification of Category I instrumentation and electrical equipment as described by General Design Criteria (GDC) 1, 2, and 23 are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10 CFR Part 50 are described in Chapter 17.

The seismic Category I instrumentation and electrical equipment and their supports are qualified in accordance with the procedures and documentation requirements of IEEE 344-1971. The recommendations of Regulatory Guide 1.100 are not applicable to Catawba based on the implementation date of the guide. However, methods similar to those described in IEEE 344-1975 are used in the qualification of instrumentation and electrical equipment as detailed in Section 3.10.2.1 and 3.10.2.2.

The seismic acceleration levels used in the seismic qualification tests and analyses are selected to envelope the plant specific levels defined in Section 3.7.

3.10.1.2 Acceptance Criteria

Seismic qualification must demonstrate that Category I instrumentation and electrical equipment is capable of performing designated safety-related functions during and after an earthquake of magnitude up to and including the Safe Shutdown Earthquake (SSE). Any spurious actuation must not result in consequences adverse to safety. The qualification must also demonstrate the structural integrity of mechanical supports and structures at the OBE level. Some permanent mechanical deformation of supports and structures is acceptable at the SSE level provided that the ability to perform the designated safety-related functions is not impaired.

3.10.2 Methods and Procedures for Qualifying Instrumentation and Electrical Equipment

3.10.2.1 NSSS Equipment

The seismic qualification of safety-related electrical equipment within the NSSS scope of supply is demonstrated by testing, analysis, or a combination of these methods in accordance with IEEE 344-1971. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including; practicability, complexity of equipment, economics, and availability of previous seismic qualification. The qualification method employed for a particular item of equipment is identified in the Catawba Equipment DataBase (EBD).

3.10.2.1.1 Seismic Qualification by Type Test

From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE 344-1971 to seismically qualify equipment. The input form

selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in Reference 1. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Reference 2). This retesting was performed at the request of the NRC Staff on agreed selected items of equipment employing multi-frequency, multi-axis test inputs (Reference 3) to demonstrate the conservatism of the original sine beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975. The original single axis sine beat testing (Reference 4) and the additional retesting completed under the Supplemental Test Program has been the subject of generic review by the NRC Staff.

3.10.2.1.2 Seismic Qualification by Analysis

The structural integrity of safety-related motors is demonstrated by a static seismic analysis in accordance with IEEE 344-1971. Should analysis fail to show the resonant frequency to be significantly greater than 33 HZ, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) calculated values are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas; 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slope at the bearings, 4) maximum stresses in the stator core welds, 5) maximum stresses in the stator core to frame welds, 6) maximum stresses in the motor mounting bolts and, 7) maximum stresses in the motor feet.

The analytical models employed and the results of the analysis are described in the qualification reference.

3.10.2.2 Balance of Plant Equipment

The seismic qualification of safety-related electrical equipment within Duke's scope of supply is demonstrated by testing, analysis, or a combination of these methods. When testing is performed, the input excitation normally employed is random, multifrequency biaxial similar to that described in IEEE 344-1975. In the event other input excitations are employed, they are selected in accordance with the requirements of IEEE 344-1971. Satisfactory operation of seismic Category I instrumentation and electrical equipment is verified both during and after testing. When testing is not practical, qualification is by analysis.

When seismic qualification of instrumentation and electrical equipment and supports is verified by testing, the test response spectrum (TRS) envelopes the required response spectrum (RRS) for all frequencies from 1 to 33 HZ. All equipment that is tested is verified to be operational during and after the test.

When seismic qualification of equipment and supports is by analysis, all stresses in critical members are verified to be less than the applicable allowable stresses. Seismically sensitive components located in the analyzed equipment are seismically qualified by testing for the localized motion at the component mounting location.

The method of qualification for each item of equipment is identified in the Catawba Equipment DataBase (EDB).

3.10.3 Methods and Procedures for Qualifying Supports of Instrumentation and Electrical Equipment

The seismic qualification of the supports for Category I electrical equipment is demonstrated by testing, analysis, or a combination of these methods. The preferred method of qualification for these supports is to test the support with the equipment as described above. When testing is not practical, qualification of supports is by static and/or dynamic analysis depending on the natural frequency of the support. In both testing and analysis procedures, the possible amplified design loads for vendor supplied equipment are considered as follows:

1. The support is tested with the actual components mounted or with the component loads simulated.
2. Analysis of the support includes the component loads.

3.10.4 Results of Tests and Analyses

The results of the seismic tests and analyses that ensure the criteria established in Section 3.10.2.1 have been satisfied, employing the qualification methods described in Sections 3.10.1 and 3.10.3, are provided in individual seismic qualification reports. These reports are referenced in the Catawba Equipment DabaBase (EDB).

3.10.5 References

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

1. Morrone, A., "Seismic Vibration Testing with Sine Beats," WCAP-7558, October 1971.
2. NS-CE-692, Letter dated July 10, 1975 from C. Eicheldinger (Westinghouse) to D. B. Vasello (NRC).
3. Jarecki, S. J., "General Method of Developing Multi-frequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary), September 1975 and WCAP-8695 (Non-Proprietary), August 1975.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

4. *Vogeding, E. L., et al, "Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP-7397-L (Proprietary) and WCAP-7817 (Non-Proprietary), December 1971 plus Supplements 1-8.*

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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 CFR 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 was originally identified in the Catawba NUREG 0588 response (Reference 1). Electrical Equipment within the scope of 10CFR50.49 is identified in the Catawba Equipment DataBase (EDB). The EDB and the Equipment Qualification Maintenance Manual (EQMM) in conjunction with administrative controls of the EQ Program, in PD-EG-ALL-1612 and AD-EG-ALL-1612, provide the documentation required under 10CFR50.49.

3.11.1.1.2 Mechanical Equipment

Mechanical equipment is assigned to seismic classifications, quality groups, and safety classes as outlined Section 3.2.2. Continued compliance with GDCs 1 and 4 for mechanical equipment is assured by means of administrative procedures for modifications, procurement, and test documentation, which require environmental qualification in a manner similar to that used for electrical equipment.

3.11.1.2 Environmental Conditions

The Catawba Environment Qualification Criteria Manual (EQCM), CNLT-1780-03.03, contain 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 6-7.

Temperature (Lower Compartment Average): Time history as shown in Figure 6-20.

Temperature (Break Compartment): Time history as shown in Figure 6-21.

Pressure (Upper and Lower Compartment): Time history as shown in Figure 6-6.

Note: The worse-case lower compartment and break compartment temperatures and the worst-case pressures 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 Sections 3.11.5, Section 12.2.1, and 12.3.2.

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 6.1.1.2 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: 171°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 Sections 3.11.5, 12.2.1, and 12.3.2.

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 Catawba Environment Qualification Criteria Manual (EQCM), CNLT-1780-03.03.

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 Sections 3.11.5.2 and 12.3.2.2. The resulting radiation environments are documented in Catawba Environment Qualification Criteria Manual (EQCM), CNLT-1780-03.03.

3.11.2 Qualification Tests and Analyses

The Catawba 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 with 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 in accordance with IEEE-323-1974.

The Environmental Qualification Maintenance Manual (EQMM-1393.01) 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.

3.11.2.1 Qualification Criteria and Standards

3.11.2.1.1 10CFR 50 Appendix A, General Design Criteria

The implementation of the General Design Criteria (GDC) is discussed in Section 3.1.

3.11.2.1.2 10CFR 50 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 10CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.

The NRC has established in 10CFR 50.49 the specific requirements that 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 PD-EG-ALL-1612 and AD-EG-ALL-1612. Additionally, The Catawba NUREG 0588 response (Reference 1) contains a discussion of compliance with 10CFR 50.49(b) and, consistent with 10CFR 50.49(k), the Duke Energy position with respect to the Category II (IEEE 323-1971) guidelines of NUREG 0588.

3.11.2.1.4 NRC Regulatory Guides

Regulatory Guide 1.30 (Safety Guide 30)

The quality assurance requirements for the installation, inspection, and testing of Class 1E electrical equipment are discussed in Chapter 17.

Regulatory Guide 1.40 (Revision 0)

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.

Regulatory Guide 1.63 (Revision 0)

Catawba 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.

Regulatory Guide 1.73 (Revision 1 5/77 supplemented 2/78)

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 (Revision 0)

The recommendations of Regulatory Guide 1.89 are not applicable to Catawba based on the implementation date of the guide.

Regulatory Guide 1.131 (August 1977)

The recommendations of Regulatory Guide 1.131 are not applicable to Catawba 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 Catawba NUREG 0588 response (Reference 1) and/or the EQMM. These qualification results have been reviewed with respect to the Duke position on the Category II (IEEE 323-1971) guidelines of NUREG 0588.

3.11.4 Loss of Ventilation

The environmental conditions in the control complex areas are maintained by redundant safety-related ventilation systems as described in Section 9.4.1. The control complex areas include the control room, cable room, battery and equipment room, switchgear rooms, motor control center rooms, ventilation equipment room, and penetration room (EL 594). Each of the safety-related ventilation systems serving these areas is designed to maintain a normal environment in the areas.

Duke Energy has analyzed the above areas assuming only one of the two redundant ventilation systems in operation. The results of this analysis are presented in Table 3-107. Class 1E equipment located in these areas is designed to perform its function within the temperature and humidity ranges shown in Table 3-107.

3.11.5 Chemical and Radiation Environment

3.11.5.1 Chemical Environment

Reference Section 6.1.1.2 for Containment chemical conditions.

3.11.5.2 Radiation Environment

The basis for determining the radiation environment for normal operation is described in Sections 12.2.1 and 12.3.2 from which generic environmental conditions are determined. Specific radiation environments are contained in the Catawba Environmental Qualification Criteria Manual (EQCM), Section 5.0.

3.11.6 References

1. Duke Power Company - Catawba Nuclear Station - Response to NUREG 0588, CNLT-1780-03.02.
2. Catawba Environment Qualification Criteria Manual (EQCM), CNLT-1780-03.03.
3. PD-EG-ALL-1612, Environmental Qualification (EQ) Program Description.

4. AD-EG-ALL-1612, Environmental Qualification (EQ) Program.

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3.12 CNS Response to Beyond-Design-Basis External Event, Fukushima Related Required Actions

3.12.1 Introduction

On March 11, 2011, an earthquake-induced tsunami caused Beyond-Design-Basis (BDB) flooding at the Fukushima Dai-ichi Nuclear Power Station in Japan. The flooding caused by the tsunami rendered the emergency power supplies and electrical distribution systems inoperable resulting in an extended loss of alternating current (AC) power (ELAP) in five of the six units on the site. The ELAP led to the loss of core cooling as well as spent fuel pool cooling capabilities and a significant challenge to containment. All direct current (DC) power was lost early in the event on Units 1 & 2 and after some period of time at the other units. Units 1, 2, and 3 were affected to such an extent that core damage occurred and radioactive material was released to the surrounding environment.

The US Nuclear Regulatory Commission (NRC) assembled a special task force, the Near-Term Task Force (NTTF) in order to advise the Commission on actions the US Nuclear Industry should undertake in order to preclude a release of radioactive material in response to a natural disaster such as that seen at Fukushima Dai-ichi. NTTF members created NRC Report "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," referred to as the "90-day Report," which contained a large number of recommendations for improving safety at US nuclear power sites.

Subsequently, the NRC issued Order EA-12-049, "Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (Agencywide Documents Access Management System (ADAMS) Package Accession No. ML12054A735) (Reference 1), Order EA-12-051, "Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (ADAMS Package Accession No. ML12054A679) (Reference 2) to implement strategies for Beyond-Design-Basis External Events (BDBEE), and reliable spent fuel pool instrumentation, respectively.

3.12.2 Order EA-12-049

NRC Order EA-12-049 was effective immediately and directed Catawba Nuclear Station to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling in the event of a beyond-design-basis external event.

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement the strategies specified in NRC Order EA-12-049. These guidelines were published in the NEI 12-06 document entitled "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (ML12242A378)(Reference 3). This guideline was endorsed by the NRC in final Interim Staff Guidance (ISG) document JLD-ISG-2012-01, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design- Basis External Events, Revision 0, dated August 29, 2012 (ML12229A174) (Reference 4).

The NEI 12-06 FLEX implementation guide adopts a three-phase approach for coping with a BDB event.

Phase 1 – The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling capabilities.

Phase 2 – The transition phase requires providing sufficient portable onsite equipment to maintain or restore these functions until resources can be brought from off site.

Phase 3 – The final phase requires obtaining sufficient offsite resources to sustain these functions indefinitely.

This three-phase approach was utilized to develop the FLEX strategies for Catawba Nuclear Station.

3.12.3 Order EA-12-051

NRC Order EA-12-051 was effective immediately and directed Catawba Nuclear Station to provide a reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. The technical scope of the NRC Order EA-12-051 specifies that Catawba Nuclear Station provide:

- Primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool
- Display in an area accessible following a severe event
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability.

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement the instrumentation specified in NRC Order EA-12-051. These guidelines were published in the NEI 12-02, Revision 1 document entitled "Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," (ML12240A307) (Reference 5). This guideline was endorsed by the NRC in final interim staff guidance (ISG) document JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0, dated August 29, 2012 (ML12221A339) (Reference 6).

The three critical levels to be monitored in the Catawba Nuclear Station spent fuel pools in which reliable indication of the water level capable of supporting identification of pool water level conditions by trained personnel are:

1. Level that is adequate to support operation of the normal fuel pool cooling system,
2. Level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and
3. Level where fuel remains covered and actions to implement make-up water addition should no longer be deferred

These three critical levels are monitored at Catawba Nuclear Station using installed Primary and Backup Spent Fuel Pool Level Instrumentation. The FLEX program document, as mentioned above, describes Spent Fuel Pool Instrumentation program requirements including procedures, testing and calibration, and quality assurance.

3.12.4 Beyond Design Basis Program

Strategies, equipment details, storage locations, periodic maintenance, and programmatic controls for mitigating beyond-design-basis external events are contained in an overall program document for flexible response to extended loss of all AC power (FLEX). Program changes are controlled in accordance with NEI 12-06, Section 11.8 (Revision 0), as endorsed by the NRC.

3.12.5 References

1. Order EA-12-049, Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, dated March 12, 2012 (ML12054A735).
2. Order EA-12-051, Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation, dated March 12, 2012 (ML12054A679).
3. NEI 12-06, Revision 0, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, dated August 2012 (ML12242A378).
4. NRC Interim Staff Guidance JLD-ISG-2012-01, Revision 0, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, dated August 29, 2012 (ML12229A174).
5. NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", dated August 2012 (ML12240A307).
6. NRC Interim Staff Guidance JLD-ISG-2012-03, Revision 0, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, dated August 29, 2012 (ML12221A339).

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