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November 27, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. B. J. Youngblood, Project Director
PWR Project Directorate No. 4

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413
Pipe Break Criteria Relief for
Reactor Coolant Loop

Reference:
1) Letter from W. H. Owen (Duke Power Company) to W. J. Dircks (NRC), dated September 19, 1983

2) Letter from H. R. Denton (NRC) to W. H. Owen (Duke Power Company), dated October 17, 1983

3) NRC Generic Letter 84-04, dated February 1, 1984

4) Letter from H. B. Tucker (Duke Power Company) to H. R. Denton (NRC) dated May 11, 1984

5) Letter from E. G. Adensam (NRC) to H. B. Tucker (Duke Power Company) dated April 23, 1985.

Dear Mr. Denton:

Please find attached a request for an exemption (pursuant to 10 CFR 50.12(a)) from General Design Criteria 4 to apply the "leak-before-break" concept to the Catawba Nuclear Station Unit 1 to eliminate postulated pipe breaks in the RCS primary loop from the plant structural design basis. Reference 1 informed the NRC that Duke Power Company was evaluating the technical feasibility and potential benefits of eliminating postulated pipe breaks in the Reactor Coolant System (RCS) primary loop from the structural design basis of the Catawba Nuclear Station. In Reference 4, Duke Power Company applied for application of Leak-Before-Break for main Reactor Coolant Loop Piping on Catawba Unit 2; a favorable ruling was made by the Commission in the subsequent Safety Evaluation Report (SER) (Reference 5). As a result of efforts by Westinghouse, the NRC, and Duke Power, Duke has concluded that it is technically feasible to eliminate these postulated pipe breaks. In addition, Westinghouse has assured Duke Power Company that the

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generic information previously submitted to the NRC to justify the elimination of RCS primary loop pipe breaks is applicable to the Catawba Nuclear Station Unit 1. Further, a safety balance in terms of accident risk avoidance versus safety gain will be demonstrated. As a result of the preceding developments, and in accordance with statements in References 2 and 3 that applications related to the leak-before-break pipe failure concept will be permitted prior to the NRC completing all of the changes in regulatory requirements, this exemption request (Attachment 1) is submitted.

The Westinghouse technical report (WCAP-10546 entitled "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Catawba Units 1 & 2" was included as Enclosure A to Reference 4 to provide technical justification for elimination of RCS primary loop breaks for Catawba. A non-proprietary version of the Specific Plant Applicability Report (WCAP-10547) was also included as enclosure B to Reference 4.

The impact on important design aspects of implementing leak-before-break on Catawba Nuclear Station Unit 1 has been evaluated by Duke Power and is summarized in Attachment 2. A detailed list of affected pipe whip restraints is provided in Attachment 3. A summary of the potential benefits which can be realised specifically from the elimination of these pipe breaks for Catawba is provided in Attachment 4. Note that these benefits total at least $813,000 and involve an estimated 397 person-rem dose reduction over the life of the Unit. Implementation of the leak-before-break concept will therefore be cost-effective as well as being technically justifiable, while resulting in improved overall plant safety. Attachment 5 consists of the revised Catawba FSAR pages associated with the elimination of RCS primary loop breaks. Once approved, these changes will be incorporated into the next applicable annual update to the Catawba FSAR.

Pursuant to 10 CFR 170.21, please find enclosed a check in the amount of $150.00.

It is requested that a resolution concerning implementation of this exemption on Catawba Unit 1 be prior to February 28, 1986. If there are any questions concerning this request, please advise.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

ROS:slb

Attachment

cc: (w/attachments, Enclosure 4)
Dr. J. Nelson Grace, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
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Dr. K. N. Jabbour
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Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Senior Resident Inspector
Catawba Nuclear Station

Attachment 1

Duke Power Company
Catawba Nuclear Station
RCS Loop Pipe Break Design Basis Exemption

Exemption Request

Pursuant to 10 CFR 50.12(a), Duke Power Company hereby applies for an exemption from the provisions to 10 CFR Part 50, Appendix A, authorizing alternative pipe break analyses utilized in resolution of generic issue A-2, "Asymmetric Blowdown Loads on PWR Primary System." The requested exemption is based upon the application of advanced fracture mechanics technology as evaluated in the Westinghouse technical report WCAP-10546.

Specifically, Duke Power Company requests the elimination of postulated circumferential and longitudinal pipe breaks in the reactor coolant system primary loop from consideration in the structural design basis of Catawba Nuclear Station Unit 1. The pipe breaks are those identified in Westinghouse topical report WCAP 8172 for the RCS primary loop. The impact on important design aspects of implementing leak-before-break on Catawba Nuclear Station Unit 1 has been evaluated by Duke Power and is summarized in Attachment 2. A detailed list of affected pipe whip restraints is provided in Attachment 3.

The bases for the requested exemption are as follow:

1. Extensive operating experience has demonstrated the integrity of the RCS primary loop including the fact that there has never been a leakage crack.

2. In-shop, pre-service, and in-service inspections performed on piping for the Catawba Nuclear Station minimize the possibility of flaws existing in such piping. The application of advanced fracture mechanics has demonstrated that if such flaws exist, they will not grow to a leakage crack when subjected to the worst case loading condition over the life of the plant.

3. If one postulates a through-wall crack, large margins against unstable crack extension exist for certain stainless steel PWR primary coolant piping when subjected to the worst case loading conditions over the life of the plant.

The application of advanced fracture mechanics technology has demonstrated that small flaws or leakage cracks (postulated or real) will remain stable and will be detected either by in-service inspection or by leakage monitoring systems long before such flaws can grow to critical sizes which otherwise could lead to large break areas such as the double-ended rupture of the largest pipe of the Reactor Coolant System. To date, use of this advanced fracture mechanics technology has been limited by the definition of a LOCA in Appendix A to 10CFR Part 50 as including postulated double-ended ruptures of piping, regardless of the associated probability. Application of the LOCA definition without regard to this advanced technology to large diameter thick-walled piping, such as the

primary coolant pipes of a PWR, imposes a severe penalty in terms of cost and occupational exposure because of the massive pipe whip restraints it requires which must be removed for in-service inspection. This penalty is unreasonable because these pipes do not have a history of failing or cracking and are conservatively designed. Accordingly, for design purposes associated with protection against dynamic effects, Duke requests this exemption from the regulations to eliminate the need to postulate circumferential and longitudinal pipe breaks.

Implementation of the exemption will have the following effects on the structural design for Catawba Nuclear Station:

1. Eliminate the need to postulate circumferential and longitudinal pipe breaks in the RCS primary loop (hot leg, cold leg, and cross-over leg piping).

2. Eliminate the need for associated pipe whip restraints in the RCS primary loop and eliminate the requirement to design for the structural effects associated with RCS primary loop pipe breaks including jet impingement.

3. Eliminate the need to consider dynamic effects and loading conditions associated with previously postulated primary loop pipe breaks. These effects include blowdown loads, jet impingement loads, and reactor cavity and subcompartment pressurization.

The exemption would not eliminate pipe breaks in the RCS primary loop as a design basis for the following:

1. Containment design
2. Sizing of Emergency Core Cooling System
3. Environmental qualification of equipment
4. Supports for heavy components

The crack sizes from the leak-before-break analysis will be used when revising reactor cavity and subcompartment pressurization data.

As stated above, Duke requests that the exemption authorize, with respect to the plant structural design basis, the elimination of pipe breaks in the RCS primary loop. Thus, the use of advanced fracture mechanics permits a deterministic evaluation of the stability of postulated flaws/leakage cracks in piping as an alternative to the current mandate of overly conservative postulations of piping ruptures. This exemption request is consistent with the provisions of footnote 1 to 10 CFR Part 50, Appendix A, which refers to the development of "further details relating to the type, size and orientation of postulated breaks in specific components of the reactor coolant pressure boundary."

As support for this request, in addition to the previously specified information, Duke would request consideration of the following:

1. Letter from Darrell G. Eisenhut (NRC) to E. P. Rahe (Westinghouse) dated February 1, 1984.

2. Memorandum from Darrell G. Eisenhut (NRC) to All Operating PWR Licensees, Construction Permit Holders and Applicants for Construction Permits, dated February 1, 1984 - Subject: Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loop (Generic Letter 84-04).

3. CRGR resolution of generic issue A-2

4. ACRS letter dated June 14, 1983, re: "Fracture Mechanics Approach to Pipe Failure."

5. Memorandum from William J. Dircks, EDO, to ACRS dated July 29, 1983, Re: "Fracture Mechanics Approach to Postulated Pipe Failures."

6. Memorandum from Harold Denton (NRC) to Murray Edelman (AIF), dated May 2, 1983.

Safety Balance

Further, pursuant to 10 CFR 50.12(a), Duke believes the requested exemption will not endanger life or property, or the common defense and security, and is in the public interest. The estimated increase in public accident exposure associated with omitting the RCS primary loop pipe whip restraints is only 0.6 person-rem per unit. This nominal estimate is based on the "Leak-before-Break Value-Impact Analysis" of Enclosure 2 to NRC Generic Letter 84-04, with adjustments made for the 40-year life and four loop design of Catawba Nuclear Station. While the value-impact analysis data is based on a population density of 340 people per square mile, projected population densities within a 50 mile radius of the Catawba site for years 1990 and 2020 are 197 and 225 people per square mile, respectively. Therefore, the 0.6 person-rem risk to public health is conservative due to lower population densities for the Catawba-specific case. After adjusting NRC Generic Letter 84-04 data for a 40-year plant life, the estimated increase in occupational accident exposure associated with omitting the RCS primary loop restraints is also low--less than 0.16 person-rem per unit for the nominal case.

The net benefit in avoidance of exposures for Catawba Unit 1 associated with the requested exemption is 397 person-rem of occupational exposure over plant life, based on Duke Power studies. This eliminated radiation exposure is related to pipe whip restraint inspection tasks, restraint disassembly/reassembly for pipe weld inspections, improved personnel access for operation and maintenance. Consequently, the savings in exposure by granting the exemption far exceed the potentially small increase in public risk and avoided accident exposure associated with omitting pipe restraint devices. Duke Power Company estimates the net cost savings for Catawba Nuclear Station Unit 1 of at least 813 thousand dollars, as presented in Attachment 4. The above net benefits have considered the costs associated with disposal of the restraint devices.

Additionally, with removal of pipe restraint devices, a substantial improvement in the quality of in-service inspections is anticipated. Also, simplified plant designs will result since removal of these restraints will eliminate potential interferences with other plant structures. Reduced RCS heat loss to containment at whip restraint locations will result. The risks of unanticipated

pipe restraint for thermal growth and seismic movement can be avoided. Thus, the exemption will lead to an overall improvement in plant safety.

With these operational benefits and with a net reduction of radiation exposure of 397 person-rem, a net safety gain has been demonstrated for Catawba Nuclear Station Unit 1. Also, a cost savings of a least 813 thousand dollars has been shown, and a technical basis for elimination of RCS primary loop pipe breaks has been demonstrated. Therefore, Duke Power Company hereby requests NRC approval of an exemption to GDC-4 in order to apply the "leak-before-break" concept to Catawba Nuclear Station Unit 1 to eliminate postulated pipe breaks in the RCS primary loop from the plant structural design basis.

ATTACHMENT 2

Impact of Elimination of Postulated Circumferential and Longitudinal Pipe Breaks in the RCS Primary Loop

STRUCTURES, SYSTEMS, COMPONENTS, PROGRAM CONSIDERED FOR IMPACT	IMPACT
Primary Loop Pipe Whip Restraints	Deleted from Design
Reactor Cavity/Primary Shield Wall/ Crane Wall/Operating Floor	Reduction in pressurization loading
Steam Generator Sub-compartment	No change
RCS Component Supports/Heavy Component Supports	No change
Emergency Core Cooling Systems	No change
Containment Design	No change
RCS Pressure Boundary Leakage	No change
Environmental Qualification Program	No change

ATTACHMENT 3

Postulated RCS Primary Loop Pipe Breaks and
Associated Pipe Whip Restraints Per Unit

Postulated Break Locations Per Loop	Associated Whip Restraint for Primary Loading
1. Reactor vessel inlet nozzle	1. Cold Leg Nozzle Break Restraint (wagon wheel)
2. Reactor vessel outlet nozzle	2. Hot Leg Nozzle Break Restraint (wagon wheel)
3. Steam generator inlet nozzle	3. Hot leg pipe whip restraint
4. 50° elbow in the intrados (longitudinal slot)	4. Hot leg pipe whip restraint
5. Steam generator outlet nozzle	5. Crossover leg pipe whip restraint (vertical run) Crossover leg elbow restraints
6. Reactor coolant pump inlet nozzle (pump suction)	6. Crossover leg elbow restraints
7. Crossover leg closure weld	7. Crossover leg elbow restraints
8. Reactor coolant pump outlet	8. None

ATTACHMENT 4

Summary of Benefits from the Elimination of Primary Loop Pipe Breaks on Catawba Nuclear Station, Unit 1

Category	Benefits
1. Plant Design	Simplifies plant design by elimination of potential interferences with piping, hangers, impulse tubing, etc.
2. Relief of congestion improving access for operation and maintenance.	397 person-rem reduction in radiation exposure over life of plant ($813K).
3. Reduction in RCS heat loss to containment at whip restraint location.	Not quantitatively accessed. Insulation can be installed on piping at current locations of RCS pipe whip restraints.
4. Improvement in overall plant safety (NUREG/CR-2136)	Improvement in ISI quality. Elimination of potential for restricted thermal or seismic movement.
5. Simplification of analysis associated with dynamic effects and loading conditions.	Pressurization loadings reduced on primary shield wall, cranewall, operating floor, and subcompartment analysis.

ATTACHMENT 5

Revision to Catawba FSAR

For Leak-Before-Break Criteria Change

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, carbon dioxide or halon 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.

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 postulated 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: Chapters 2.0, 3.0 and 6.0.

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

Each rod cluster control assembly is provided with a sensor to detect positioning at the bottom of its travel. This condition is also alarmed in the Control Room. Four ex-core long ion chambers also detect asymmetrical flux distributions indicative of rod misalignment.

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

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

Reference: Chapters 7.0 and 11.0.

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, or rapidly propagating failure, and of gross rupture.

DISCUSSION:

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

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

General Design Criterion 4 of Appendix A to 10CFR50 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
MEB at these locations will not result in unacceptable consequences to essential
Q102 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.

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. The safety evaluation of each interaction is described in Sections 3.6.1.3 and 3.6.1.1.5.

a) 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.

b) 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.

c) 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 3.6.2-1) is restrained to meet the following criteria:

a) Propagation of the break to the unaffected loops is prevented to assure the delivery capacity of the accumulators and low head pumps.

b) 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 K.3 of Section 3.6.2.1.2).

c) 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 refueling cavity walls, various structural beams, the operating floor, and the crane wall enclose each reactor coolant loop in a separate compartment; thereby preventing an accident in any loop branch connection from affecting another loop or the Containment. The portion of the main steam and feedwater lines within the Containment has been routed behind barriers to separate these lines 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.

Table 3.6.1-1 provides a listing of high-energy systems. Moderate-energy systems are listed in Table 3.6.1-2. Control room habitability is discussed in Section 3.6.1.1.3.4.

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 3.6.2-1) on outgoing lines (*) and down to and including the second check valve (Case III in Figure 3.6.2-1) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve does not result in an uncontrolled loss of reactor coolant 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 3.6.2-1), a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

(*)It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operators of both valves.

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 1.a 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 3.6.2-1 and Figure 3.6.2-2.

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, 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, pipe break of these lines results in a rapid blowdown of the Reactor Coolant System and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

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

a) 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.

b) Station general arrangement and layout design of high-energy systems utilize the possible combination of physical separation, pipe bends, pipe whip restraints and encased or jacketed piping for the most practical design of the station. These possible design combinations decrease postulated piping break consequences to minimum and acceptable levels. In all cases, the design is of a nature to mitigate the consequences of the break so that the reactor can be shutdown safely and eventually maintained in a cold shutdown condition.

c) 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.

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:

a) High-Energy Piping

MEB Q23

1) Circumferential Breaks and Longitudinal Splits
   a) Pipe Whip (displacement)
   b) Jet Impingement
   c) Compartment Pressurization
   d) Flooding
   e) Environmental Effects (Temperature, humidity, water spray)

2) Throughwall leakage cracks
   a) Environmental Effects (Temperature, Humidity)
   b) Flooding

b) Moderate-Energy Piping

MEB Q23

1) Through-wall leakage cracks
   a) Flooding
   b) Environmental Effects (Temperature, humidity, water spray)
   c) Water Spray

For high energy piping there are certain exceptions as detailed in Reference 1a 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

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 connections. Further discussion of the dynamic analysis methods used to verify the design adequacy of the reactor coolant loop piping, equipment and supports is given in 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 unac-ceptable 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 3.6.2-4.

The steps in the analytical method are:

a) The initial deflected position of the Reactor Coolant System model is defined by applying the general pressure analysis;

b) Natural frequencies and normal modes of the broken branch connection are determined;

c) 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;

d) 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

e) 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 connections. The general dynamic solution process is shown in Figure 3.6.2-5.

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 cal-culation 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.

REFERENCES FOR SECTION 3.6

1. "Pipe Breaks for the Loca Analysis of the Westinghouse Primary Coolant Loop", WCAP-8082-P-A, January, 1975 (Proprietary) and WCAP-8172-A (Non-Proprietary), January, 1975.

1.a. Letter from H. B. Tucker (DPC) to ~~H. R. Denton~~ EG Adensam (NRC), dated ~~December 20, 1983~~ May 11, 1984, transmitting Westinghouse report justifying elimination of RCS primary loop pipe breaks ~~fro~~ from certain design considerations.

2. "Documentation of Selected Westinghouse Structrual Analysis Computer Codes", WCAP-8252, Revision 1, May, 1977.

3. Bordelon, F.M., "A Comprehensive Space-Time Dependent Analysis of Loss of Coolant (SATAN IV Digital Code)", WCAP-7263, August, 1971 (Proprietary) and WCAP-7750, August, 1971 (Non-Proprietary).

4. American Institute for Steel Construction, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings", Februrary 12, 1969.

*Table 3.6.1-3 (Page 1)
Comparison of Duke Pipe Rupture Criteria And
NRC Requirements of Branch Technical Positions
APCSB 3-1 (November 1975), MEB 3-1 (November 1975), and NRC Regulatory Guide 1.46 (May 1973)

NRC Criteria	Duke Criteria
APCSB 3-1, Section B.2.c	SAR Section 3.6.2
Section B.2.c. requires that piping between containment isolation valves be provided with pipe whip restraints capable of resisting bending and torsional moments produced by a postulated failure either upstream or downstream of the valves. Also, the restraints should be designed to withstand the loadings from postulated failures so that neither isolation valve operability nor the leaktight integrity of the containment will be impaired.	Duke criteria is generally equivalent to NRC criteria as clarified below: The containment structural integrity is provided for all postulated pipe ruptures. In addition, for any postulated rupture classified as a loss of coolant accident, the design leaktightness of the containment fission product barrier will be maintained. Penetration design is discussed in SAR Section 3.6.2.4. This section also discussed penetration guard pipe design criteria.
Terminal ends should be considered to originate at a point adjacent to the required pipe whip restraints.	Terminal ends are defined as piping originating at structure or component that act as rigid constraint to the piping thermal expansion.
APCSB 3-1, Section B.2.d	SAR Section 6.6
(1) The protective measures, structures, and guard pipes should not prevent the access required to conduct inservice inspection examination. (2) For portions of piping between containment isolation valves, the extent of inservice examinations completed during each inspection interval should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds.	Duke criteria is different than the NRC criteria due to the code effective date as described below: ASME Class 2 piping welds will be inspected in accordance with requirements given in SAR Section 6.6.

*Pipe breaks in the RCS primarly loop are not postulated for consideration in certain aspects of plant design, as defined in Reference 1a.

Table 3.6.2-1

Postulated Break Locations For The Main Coolant Loop

Location of Postulated Rupture	Type
*1. Reactor Vessel Outlet Nozzle	Circumferential
*2. Reactor Vessel Inlet Nozzle	Circumferential
*3. Steam Generator Inlet Nozzle	Circumferential
*4. Steam Generator Outlet Nozzle	Circumferential
*5. Reactor Coolant Pump Inlet Nozzle	Circumferential
*6. Reactor Coolant Pump Outlet Nozzle	Circumferential
*7. 50° Elbow on the Intrados	Longitudinal
*8. Loop Closure Weld in Crossover Leg	Circumferential
9. Residual Heat Removal (RHR) Line/Primary Coolant Loop Connection	Circumferential (Viewed from the RHR line)
10. Accumulator (ACC) Line/Primary Coolant Loop Connection	Circumferential (Viewed from ACC line)
11. Pressurizer Surge (PS) Line/Primary Coolant Loop Connection	Circumferential (Viewed from the PS line)

*Reference 1 defines the original basis for postulating pipe breaks in the reactor coolant system primary loop. Reference 1a provides the basis for eliminating this previously postulated pipe break from certain aspects of design consideration.

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

are increased by 40 percent for design purposes. These increased design pressures are also listed in Table 3.8.3-2.

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 3.8.3-3 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 3.8.3-2 and 3.8.3-3 respectively, are not applicable as listed but represent an upper bound for loadings resulting from a postulaed pipe break. The final differential 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 conditons. 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 conditons 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.

Q 220.44 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 3.8.3-4.

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 enclo-

sure 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 (A and B on Figure 3.8.3-4) of the Structural Section 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 Figures 3.8.3-5 and 3.8.3-6. A comparison of the results is shown in Figures 3.8.3-7 through 3.8.3-8. 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 Section 3.7.2 and Section 3.8.2.4 for description of program.

2. For the stresses, stress resultants and displacements of a shell of revolution due to the transient dynamic pressures associated with a loss-of-coolant accident: A computer program originally written at the University of California, Berkeley. Refer to Section 3.8.2.4 for description of the program.

3. For seismic response of structures that can be idealized as multi-mass systems: A computer program based on the theory presented in Section 3.7.2.1 and 3.7.2.6.

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 as shown on the graph. 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 total number of occurrences of this transient during the 40-year design 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)

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 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

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.

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 3.9.1-2. 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 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.

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 10CFR50, Section 50.55a is discussed in Sections 3.1 and 5.2 conformance with Regulatory Guides pertaining in Section 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 3.9.1-1. 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.

3.9.5.3 Design Loading Categories

The combination of design loadings fit into either the normal, upset, emergency 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 3.9.1-1.

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 3.9.2-2. The corresponding no loss of function limits are included in Table 3.9.2-2 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.

REFERENCES FOR SECTION 3.9 (cont'd)

16. Letter from H.B. Tucker (DPC) to ~~H.R. Denton~~ EG Adensam (NRC), dated ~~December 20, 1983~~ May 11, 1984, transmitting Westinghouse report justifying elimination of RCS primary loop breaks for certain design considerations.

Table 3.9.1-1 (page 2)
Design Transients for ASME Code Class 1 Piping

(1) DESIGN TRANSIENTS	(2) CONDITION	(3) OCCURRENCES	(4) RESIDUAL HEAT REMOVAL SYSTEM	SAFETY INJECTION SYSTEM	CHEMICAL AND VOLUME CONTROL SYSTEM	PRESSURIZER SURGE LINE	PRESSURIZER RELIEF	PRESSURIZER SPRAY	RTD BYPASS	REACTOR COOLANT DRAIN LINES	UPPER HEAD INJECTION LINES
Loss of Load without Immediate Turbine or Reactor Trip	Upset	80	X	X	X	X	X NOTES 4, 5	X	X	X	X
Loss of Flow in One Loop	Upset	80	X	X	X		X	X	X	X	X
Reactor Trip with Cooldown and Inadvertent SIS Actuation	Upset	10	X	X	X	X	X	X	X	X	X
Inadvertent RCS Depressurization	Upset	20	X	X	X	X	X	X	X	X	X
Inadvertent SI Accumulator Blowdown during Plant Cooldown	Upset	4	-	X	-	-	-	-	-	-	-
High Head Safety Injection	Upset	22	-	X	-	-	-	-	-	-	-
Boron Injection	Upset	48	-	X	-	-	-	-	-	-	-
Large Steam Break	Faulted	1	X	X	X	X	X	X	X	X	X
Pipe Rupture	Faulted	1	X	X	X	X	X	X	X	X	X
High Head Safety Injection	Faulted	2	-	X	-	-	-	-	-	-	-
Boron Injection	Faulted	2	-	X	-	-	-	-	-	-	-
Turbine Roll Test	Test	10	X	X	X	X	X	X	X	X	X
Hydrostatic Test	Test	5	X	X	X	X	X	X	X	X	X
Primary Side Leak Test	Test	50	X	X	X	X	X	X	X	X	X
Inadvertent Auxiliary Spray	Test	1	-	-	X	-	-	X	-	-	-

NOTES:

1. Pressurizer surge line is analyzed for 80 occurrences of transient C-7, the final cooldown spray.
2. Pressurizer surge line is analyzed for 150,000 initial fluctuations and 3,000,000 random fluctuations.
3. These transients are conditions which can cause the PORV's to open. Although a total of 320 such transients are shown, the PORV inlet lines are analyzed for 100 such occurrances.
4. For analysis of the safety valves 40 occurrences were assumed.
5. Number of occurrences is 20,000,000.

2. Analysis of Accident Loads

As shown in Reference 7, grid crushing tests and seismic and LOCA evaluations show that the fuel assembly will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event. The seismic and LOCA evaluations given in reference 7 (which encompass the Catawba plant) are conservative when compared to the Catawba plant's design bases relative to the structural integrity of the reactor coolant system (RCS primary loop). As discussed in Section 3.6, the elimination of consideration of the dynamic effects of pipe breaks in the RCS primary loop has been fully justified.

A prototype fuel assembly has been subjected to column loads in excess of those expected in normal service and faulted conditions (see Reference 7).

No interference with control rod insertion into thimble tubes will occur during a Safe Shutdown Earthquake (SSE).

Stresses in the fuel assembly caused a tripping of the rod cluster control assembly have little influence on fatigue because of the small number of events during the life of an assembly. Assembly components and prototype fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met (Reference 7).

3. Loads Applied in Fuel Handling

The fuel assembly design loads for shipping have been established at 6 g's. Accelerometers are permanently placed into the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Past history and experience has indicated that loads which exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components such as the grid assembly, sleeves, inserts and structure joints have been performed to assure that the shipping design limits do not result in impairment of fuel assembly function.

4.2.3.6 Reactivity Control Assembly and Burnable Poison Rods

1. Internal Pressure and Cladding Stresses During Normal, Transient and Accident Conditions

The designs of the burnable poison, source rods and B_4C absorber rods provide a sufficient cold void volume to accommodate the internal pressure increase during operation.

For the burnable poison rod, the use of glass in tubular form provides a central void volume along the length of the rods. For the source rods, and the B_4C absorber rod, a void volume is provided in the cladding in order to limit the internal pressure increase until end-of-life (see Figure 4.1.1-12).

The stress analysis of the burnable poison and source rods assumes 100 percent gas release to the rod void volume in addition to the initial pressure within the rod. For the B_4C control rod a 20% gas release is assumed.

5.4.14.2.2 Reactor Coolant Pump

The reactor coolant pump support system consists of vertical steel columns and a lateral steel frame. Figures 5.4.14-3 through 5.4.14-5 show outlines of the support system of the reactor coolant pump.

5.4.14.2.3 Pressurizer

The pressurizer support system consists of vertical steel hangers from the operating floor to the base of the pressurizer, a lateral frame at the base anchored to the crane wall and tied to the vertical hangers, and an upper lateral steel ring anchored to the crane wall and pressurizer enclosure walls. Figures 5.4.14-6 through 5.4.14-8 show outlines of the pressurizer support system.

5.4.14.2.4 Reactor Vessel

The reactor vessel supports are individual water-cooled rectangular box structures beneath the vessel nozzles and anchored to the primary shield wall. Figure 5.4.14-9 shows an outline of a typical reactor vessel support.

5.4.14.3 Fabrication

The fabrication of all steel component supports is in accordance with Subsection NF of Section III of the 1974 or 1977 ASME Code, depending on the contract date for the particular support. A code stamp is not required.

5.4.14.4 Materials

The materials used for all steel supports are listed in Table 5.4.14-2. For all materials except the reactor coolant pump bolts (See Figure 5.4.14-3), the materials meet the requirements of Article NF-2000 of Section III of the ASME Code. The reactor coolant pump bolt material is a high strength steel (modified 4340) not defined in Appendix I of Section III. This material is required to pass Charpy V-notch impact tests. In addition, the material is not subjected to stress corrosion cracking by virtue of the fact that a corrosive environment is not present and the bolt has essentially no residual stresses and does not experience any significant sustained loads during normal service.

Concrete support structures are constructed in accordance with the ACI Code 318-71 using grade 60 reinforcing and 5000 psi concrete.

Figures 5.4.14-10 thru -15

Deleted by Revision 9

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The containment vessel steel shell is designed for dead loads, construction loads, design basis accident loads, external pressure, seismic loads and penetration loads as described in Section 3.8.2.3. The applicable loading combinations considered are listed in Table 3.8.2-1.

The design basis accident internal pressure is 15 psig. The effects of pipe rupture in the primary coolant system up to and including a double-ended rupture of the largest pipe as well as rupture of the main steam line are considered in determining the peak accident pressure.

The maximum design external pressure is 1.5 psig. This is greater than the internal vacuum created by an accidental trip of a portion of the Containment Spray System during normal operation. The Containment Pressure Control System is discussed in Section 7.6.

The internal structures of the containment vessel are also designed for subcompartment differential accident pressures. The accident pressures considered are due to the same postulated pipe ruptures as described above for the containment vessel or as described in Section 3.6, as applicable. A 40 percent margin is applied to these calculated differential pressures. A tabulation of the calculated as well as the design pressures (including the 40 percent increase) is given in Table 3.8.3-2.

The other simultaneous loads in combination with the accident pressures and the applicable load factors are given in Table 3.8.1-2. For a further description of these loads see Section 3.8.3.7.

The functional design of the Containment is based upon the following accident input source term assumptions and conditions.

(1) The design basis blowdown energy of 324.2×10^6 Btu and mass of 498,200 lb put into the Containment.

(2) The hot metal energy is considered.

(3) A reactor core power of 3526 MWt (plus 2%) used for decay heat generation.

(4) The minimum Engineered Safety Feature performance (i.e., the single failure criterion applied to each safety system) comprised of the following:

   a. The ice condenser which condenses steam generated during a LOCA thereby limiting the pressure peak inside the Containment (see Section 6.7).

   b. The Containment Isolation System which closes those fluid penetrations not serving accident consequence limiting purposes (see Section 6.2.4).

Refer to Section 6.2.1.5 for an analysis of the minimum containment pressure transient used in the analysis of the emergency core cooling system.

ICSB Q11 Instrumentation provided to monitor and record the containment pressure during the course of an accident within the containment is discussed in Chapter 7.

Ice condenser instrumentation is discussed in Section 6.7.15.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Basis

Consideration is given in the design of the Containment internal structures to localized pressure pulses that could occur following a loss-of-coolant accident. If a loss-of-coolant accident were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a rate faster than the overall Containment, thus imposing a differential pressure across the walls of the structures.

These subcompartments include the steam generator enclosure, pressurizer enclosure, and the reactor cavity. Each compartment is designed for the largest blowdown flow resulting from the severance of the largest connecting pipe within the enclosure or the blowdown flow into the enclosure from a break in an adjacent region.

The extent to which pipe restraints are used to limit the break area of pipe ruptures is presented in Section 3.9.

The preliminary calculated differential compartment pressures are increased by a minimum of 40 percent for the design of interior structure walls, slabs, and component supports. The final calculated differential compartment pressures and component support loads due to final calculated differential pressures are in all cases less than those used for design.

The subcompartment pressurization following a loss-of-coolant accident, was considered in the design of the interior structure. Subsequent to this design a revised postulated pipe break criteria was introduced in Section 3.6. The subcompartment pressurizations resulting from loss-of-coolant accident is not applicable, as described in this section, but represent an upper bound for loadings resulting from a postulated pipe break. The final calculated differential compartment pressures and component support loads due to final calculated differential pressures are in all cases less than those used for design.

The basic performance of the Ice Condenser Reactor Containment System has been demonstrated for a wide range of conditions by the Waltz Mill Ice Condenser Test Program. These results have clearly shown the capability and reliability of the ice condenser concept to limit the Containment pressure rise subsequent to a hypothetical loss-of-coolant accident.