ENCLOSURE B

TECHNICAL BASES FOR ELIMINATING LARGE PRIMARY LOOP PIPE RUPTURES AS THE STRUCTURAL DESIGN BASIS FOR CATAWBA UNITS 1 AND 2

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1.0 INTRODUCTION

1.1 Purpose

The current structural design basis for the reactor coolant system (RCS) primary loop requires that pipe breaks be postulated as defined in the approved Westinghouse Topical Report WCAP 8082, Reference 1. In addition, protective measures for the dynamic effects associated with RCS primary loop pipe breaks have been incorporated in the Catawba plant design. However, Westinghouse has demonstrated on a generic basis that RCS primary loop pipe breaks are highly unlikely and should not be included in the structural design basis of Westinghouse plants (see Reference 2). The purpose of this report is to demonstrate that the generic evaluations performed by Westinghouse are applicable to the Catawba plant. In order to demonstrate this applicability, Westinghouse has performed a comparison of the loads and geometry for the Catawba plant with envelope parameters used in the generic analyses (Section 3.0); fracture mechanics evaluation (Section 4.0); determination of leak rates from a through-wall crack (Section 5.0), fatigue crack growth evaluation (Section 6.0); and conclusions (Section 7.0).

1.2 Scope

This report applies to the Catawba plant reactor coolant system primary loop piping. It is intended to demonstrate that specific parameters for the Catawba plant are enveloped by the generic analysis performed by Westinghouse in WCAP-9570 (Reference 3) and accepted by the NRC as noted in a letter from Harold Denton dated May 2, 1983 (Reference 4).

1.3 Objectives

The conclusions of this report (Reference 3) support the elimination of RCS primary loop pipe breaks for the Catawba plant. In order to validate this conclusion the following objectives must be achieved.

a. Demonstrate that Catawba plant parameters are enveloped by generic Westinghouse studies.

b. Demonstrate that margin exists between the critical crack size and a postulated crack which yields a detectable leak rate.

c. Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of the Catawba plant.

d. Demonstrate that fatigue crack growth is negligible.

1.4 Background Information

Westinghouse has performed considerable testing and analysis to demonstrate that RCS primary loop pipe breaks can be eliminated from the structural design basis of all Westinghouse plants. The concept of eliminating pipe breaks in the RCS primary loop was first presented to the NRC in 1978 in WCAP 9283 (Reference 5). This Topical Report employed a deterministic fracture mechanics evaluation and a probabilistic analysis to support the elimination of RCS primary loop pipe breaks.

This approach was then used as a means of addressing Generic Issue A-2 and Asymmetric LOCA Loads. Westinghouse performed additional testing and analysis to justify the elimination of RCS primary loop pipe breaks. As a result of this effort, WCAP 9570 was submitted to the NRC. The NRC evaluated the technical merits of this concept and prepared a draft SER in late 1981 endorsing this concept. Additionally, both Harold Denton and the ACRS have endorsed the technical acceptability of the Westinghouse evaluations. Specifically, in a May 2, 1983 letter (Reference 4) Harold Denton states that ". . . it is technically satisfied with Westinghouse Topical Report 9570 Rev. 2 . . . ." Additionally, the ACRS stated in a June 14, 1983 letter (Reference 6) that "... there is no known mechanism in PWR primary piping material for developing a large break without going through an extended period during which the crack would leak copiously."

The NRC funded research through Lawrence Livermore National Laboratory (LLNL) to address this same issue using a probabilistic approach. As part of the LLNL research effort, Westinghouse performed extensive evaluations of specific plant loads, material properties, transients, and system geometries to demonstrate that the analysis and testing previously performed by Westinghouse and the research performed by LLNL applied to all Westinghouse plants including Catawba (References 7 and 8). The results from the LLNL study were released at a March 28, 1983 ACRS Subcommittee meeting. These studies which are applicable to all Westinghouse plants east of the Rocky Mountains, determined the mean probability of a direct LOCA (RCS primary loop pipe break) to be 10^{-10} per reactor year and the mean probability of an indirect LOCA to be 10^{-7} per reactor year. Thus, the results previously obtained by Westinghouse (Reference 5) were confirmed by an independent NRC research study.

The above studies establish the technical acceptability for eliminating pipe breaks from the Westinghouse RCS primary loop. The LLNL study has been shown applicable to Catawba plant by inclusion of plant specific data. This report will demonstrate the applicability of the Westinghouse generic evaluations to the Catawba plant.

2.0 OPERATION AND CHEMICAL STABILITY OF THE PRIMARY COOLANT SYSTEM

The Westinghouse reactor coolant system primary loop has an operating history (over 400 reactor years) which demonstrates its inherent stability characteristics. Additionally, there is no history of cracking in RCS primary loop piping. In addition to the fracture resistant materials used in the piping system, the chemistry of the reactor coolant is tightly controlled and variations in temperatures, pressure and flow during normal operating conditions are insignificant.

As stated above, the reactor coolant chemistry is maintained within very specific limits. For example, during normal operation oxygen in the coolant is limited to less than [ ]$^{+}$ This stringent oxygen limit is achieved by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at a concentration of [ ]$^{+}$ The oxygen concentration in the reactor coolant is verified by routine sampling and chemical analysis. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides at or below [ ]$^{+}$ This concentration is assured by controlling charging flow chemistry and specifying proper wetted surface materials. Halogen concentrations are also verified by routine chemical sampling and analysis. +a,c, +a,c, +a,c,

In order to ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within [ ]$^{+}$ by control rod position. Pressure is controlled by pressurizer heaters and pressurizer spray, to a variation of less than [ ]$^{+}$ for steady state conditions. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely, system resistance and the reactor coolant pump characterisitcs are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow characteristics of the system. +a,c,e +a,c,e

The reactor coolant system, including piping and primary components, is designed for normal, upset, emergency and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity.

3.0 PIPE GEOMETRY AND LOADING

A segment of the primary coolant hot leg pipe is shown in Figure 1. This segment is postulated to contain a circumferential through-wall flaw. The inside diameter and wall thickness of the pipe are 31.0 and 2.61 inches, respectively. The pipe is subjected to a normal operating pressure of [ ]+ psia. The design calculations indicate that the junction of the [+a,c
]+ is most highly stressed. At +a,c
this location the axial load, F, and the total bending moment M, are [1849]+ +a,c
kips (including the axial force due to pressure) and [ ]+ in-kips, +a,c
respectively. Figure 2 identifies the loop weld locations. The material properties and the loads at these locations resulting from Deadweight, Thermal Expansion and Safe Shutdown Earthquake are indicated in Table 1. The method of obtaining these loads can be briefly summarized as follows:

The axial force F and transverse bending moments, M_y and M_z, are chosen for each static load (pressure, deadweight and thermal) based on elastic-static analyses for each of these load cases. These pipe load components are combined algebraically to define the equivalent pipe static loads F_s, M_{ys}, and M_{zs}. Based on elastic SSE response spectra analyses, amplified pipe seismic loads, F_d, M_{yd}, M_{zd} are obtained. The maximum pipe loads are obtained by combining the static and dynamic load components as follows:

$$F = \left| F_s \right| + \left| F_d \right|$$

$$M = \sqrt{M_y^2 + M_z^2}$$

where

$$M_y = \left| M_{ys} \right| + \left| M_{yd} \right|$$

$$M_z = \left| M_{zs} \right| + \left| M_{zd} \right|$$

The corresponding geometry and loads used in the reference report (Reference 3) are as follows: inside diameter and wall thickness are 29.0 and 2.5 inches; axial load and bending moment are [ ]+ inch kips. +a,c
The outer fiber stress for Catawba is [ ]+ ksi, while for the reference report it is [ ]+ ksi. This demonstrates conservatism in the reference report which makes it more severe than the Catawba project. +a,c +a,c

4.0 FRACTURE MECHANICS EVALUATION

4.1 Global Failure Mechanism

Determination of the conditions which lead to failure in stainless steel must be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. A conservative method for predicting the failure of ductile material is the [

]+ This methodology has been shown to be applicable to ductile piping through a large number of experiments, and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring [+a,c

]+ (Figure 3) when loads are applied. The detailed development is provided in Appendix A, for through-wall circumferential flaw in a pipe with internal pressure, axial force and imposed bending moments. The [+a,c

]+ for such a pipe is given by: +a,c

[ ] +a,c,e

[ ]+a,c,e

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect [ ]+ Good agreement was found between the analytical predictions and the experimental results [9]. +a,c,

4.2 Local Failure Mechanism

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally crack instability. Depending on the material properties and geometry of the pipe, flaw size, shape and loading, the local failure mechanisms may or may not govern the ultimate failure.

The stability will be assumed if the crack does not initiate at all. It has been accepted that the initiation toughness, measured in terms of J_{IN} from a J-integral resistance curve is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than J_{IN} of the material, then the crack will not initiate. If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation:

$$T_{app} = \frac{dJ}{da} \frac{E}{\sigma_f^2}$$

where

T_{app} = applied tearing modulus

E = modulus of elasticity

σ_f = [ ]+ (flow stress) +a,c,e

a = crack length

[σ_y,σ_u = yield and ultimate strength of the material, respectively.]+ +a,c,e

In summary, the local crack stability will be established by the two step criteria:

$$J < J_{IN}$$

$$T_{app} < T_{mat} \quad \text{if } J > J_{IN}$$

4.3 Results of Crack Stability Evaluation

Figure 4 shows a plot of the [ ]+ as a function of through-wall circumferential flaw length in the [cross-over leg]+ of the main coolant piping. This [ ]+ was calculated for Catawba data of a pressurized pipe at [ ]+ with ASME Code minimum [ ]+ properties. The maximum applied bending moment of [ ]+ in-kips can be plotted on this figure, and used to determine the critical flaw length, which is shown to be [ ]+ inches. This is considerably larger than the [ ]+ inch reference flow used in Reference 3. +a,c,

[ ] +a,c,

[

]+ Therefore, it can be concluded that a postulated [ ]+ inch through-wall flaw in the Catawba loop piping will remain stable from both a local and global stability standpoint. +a,c,e +a,c,e

5.0 LEAK RATE PREDICTIONS

Leak rate calculations were performed in Reference 3 using an initial through-wall crack [ ]+. The computed leak rate was [ ]+ based on the normal operating pressure of [ ]+ psi. [ ]+ +a,c,

This computed leak rate [ ]+ significantly exceeds the smallest detectable leak rate for the plant. The Catawba plant has a RCS pressure boundary leak detection system which is consistent with the requirements of Regulatory Guide 1.45 and can detect leakage of 1 gpm in one hour. There is a factor of [ ]+ between the calculated leak rate and the Catawba plant leak detection systems. +a,c

6.0 FATIGUE CRACK GROWTH ANALYSIS

To determine the sensitivity of the primary coolant system to the presence of small cracks, a fatigue crack growth analysis was carried out for the [ ]+ region of a typical system. This region was selected +a,c because it is typically one of the highest stressed cross sections, and crack growth calculated here will be conservative for application to the entire primary coolant system.

A finite element stress analysis was carried out for the [+a,c]+ of a plant typical in geometry and operational characteristics to any Westinghouse PWR System. [

]+ All normal, upset and test +a,c conditions were considered, and circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in three different locations, as shown in Figure 5. Specifically, these were:

Cross Section A: [
Cross Section B: +a,c,e
Cross Section C:]

Fatigue crack growth rate laws were used [

]+ The law for stainless steel was +a,c, derived from Reference 11, with a very conservative correction for R ratio, the ratio of minimum to maximum stress during a transient.

$$\frac{da}{dn} = (5.4 \times 10^{-12})\ K_{eff}^{4.48} \text{inches/cycle}$$

where $K_{eff} = K_{max}\ (1-R)^{0.5}$

$R = K_{min}/K_{max}$

[ ]+a,

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 2, and shows that the crack growth is very small, regardless [ ]+a,c

7.0 CONCLUSIONS

This report has established the applicability of the generic Westinghouse evaluations which justify the elimination of RCS primary loop pipe breaks for the Catawba plant as follows:

a. The loads, material properties, transients and geometry relative to the Catawba RCS primary loop are enveloped by the parameters of WCAP 9570.

b. The critical crack length at the worst location in the RCS primary loop is [ ]+ This is significantly greater than the [ ]+ inches stable crack used as a basis for calculating leak rates in WCAP 9570. +a, +a,

c. The leakage through a [ ]+ crack in the RCS primary loop is [ ]+ based on WCAP 9570. The Catawba plant has a RCS pressure boundary leak detection system which is consistent with the requirements of Regulatory Guide 1.45 and can detect leakage of 1 gpm in one hour. Thus, there is a factor of [ ]+ between the calculated leak rate and the Catawba plant leak detection systems. +a, +a,

d. Fatigue crack growth was determined for postulated flaws and was found to be extremely small over plant life and, therefore, is considered insignificant.

Based on the above, it is concluded that RCS primary loop pipe breaks should not be considered in the structural design basis of the Catawba plant.

8.0 REFERENCES

1. WCAP 8082 P-A, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," Class 2, January 1975.

2. Letter from Westinghouse (E. P. Rahe) to NRC (R. H. Vollmer) dated May 11, 1983.

3. WCAP 9570, Rev. 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," Class 3, June 1981.

4. Letter from NRC (H. R. Denton) to AIF (M. Edelman) dated May 2, 1983.

5. WCAP 9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," Class 2, March, 1978.

6. Letter from ACRS (J. J. Ray) to NRC (W. J. Dircks) dated June 14, 1983.

7. Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated April 25, 1983.

8. Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

9. Kanninen, M. F., et al, "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks" EPRI NP-192, September 1976.

10. Bush, A. J., Stooffer, R. B. "Fracture Toughness of Cast 316SS Piping Material Heat No. 156576, at 600°F", W R&D Memo No. 83-5P6EVMTL-M1, March 7, 1983, Westinghouse Proprietary Class 2.

11. Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment" Trans. ASME Journal of Pressure Vessel Technology Vol. 101, Feb. 1979.

[ ] +a,c,e

APPENDIX A

+a,c

TABLE 1

CATAWBA PRIMARY LOOP DATA

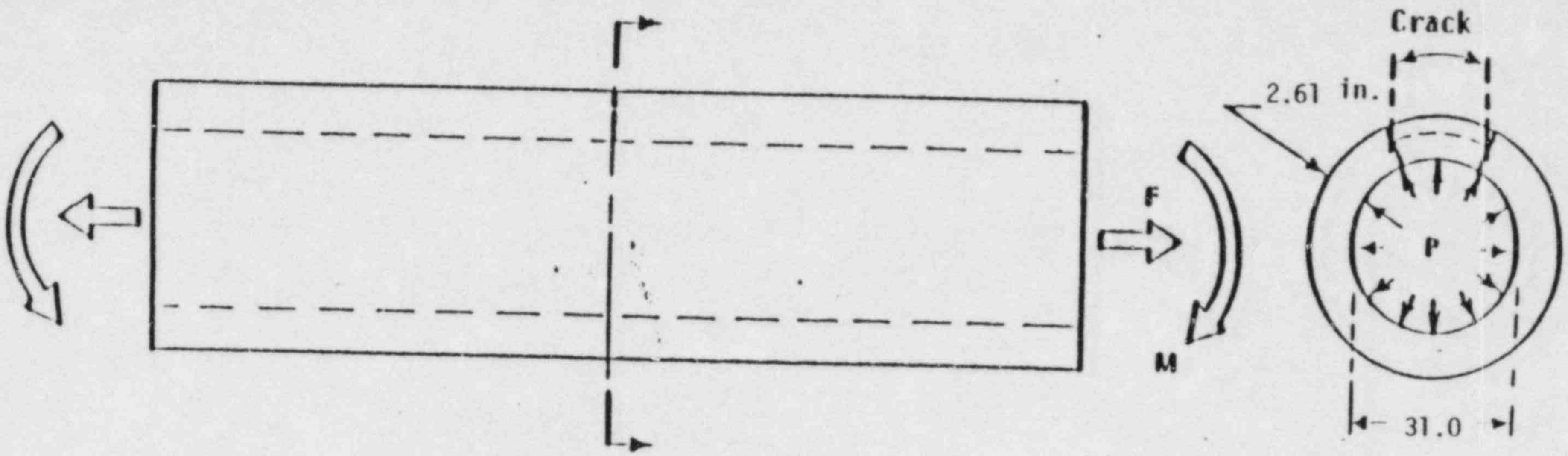
a,c,e

*The Highest Stressed Location

TABLE 2

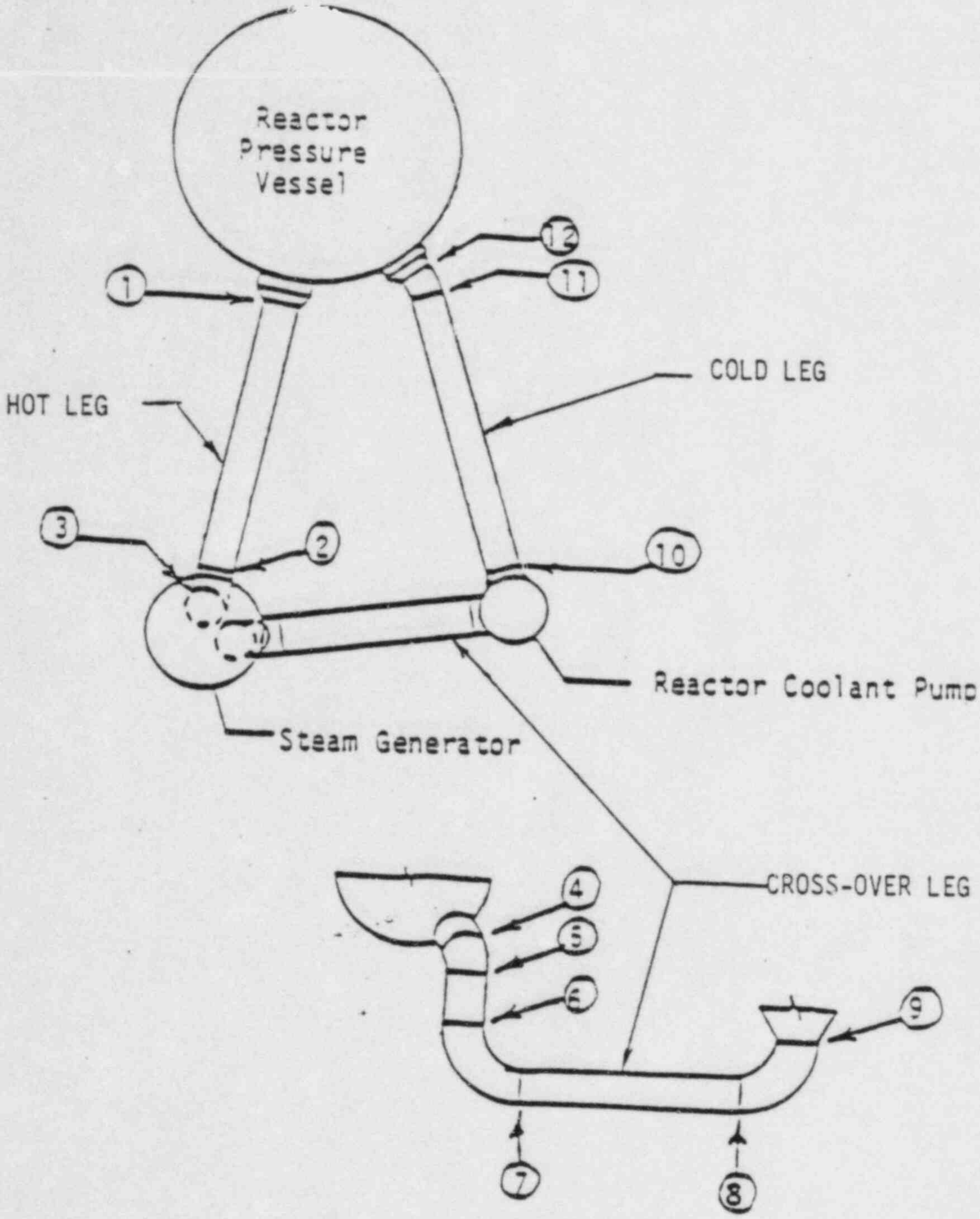
FATIGUE CRACK GROWTH AT [ ]+ (40 YEARS) +a,c,e

INITIAL FLAW (IN)	FINAL FLAW (IN)			
	[ ]+	[ ]+	[ ]+	+a,c,e
0.292	0.31097	0.30107	0.30698	
0.300	0.31949	0.30953	0.31626	
0.375	0.39940	0.38948	0.40763	
0.425	0.45271	0.4435	0.47421	



$[\quad]^{+a,c,e}$

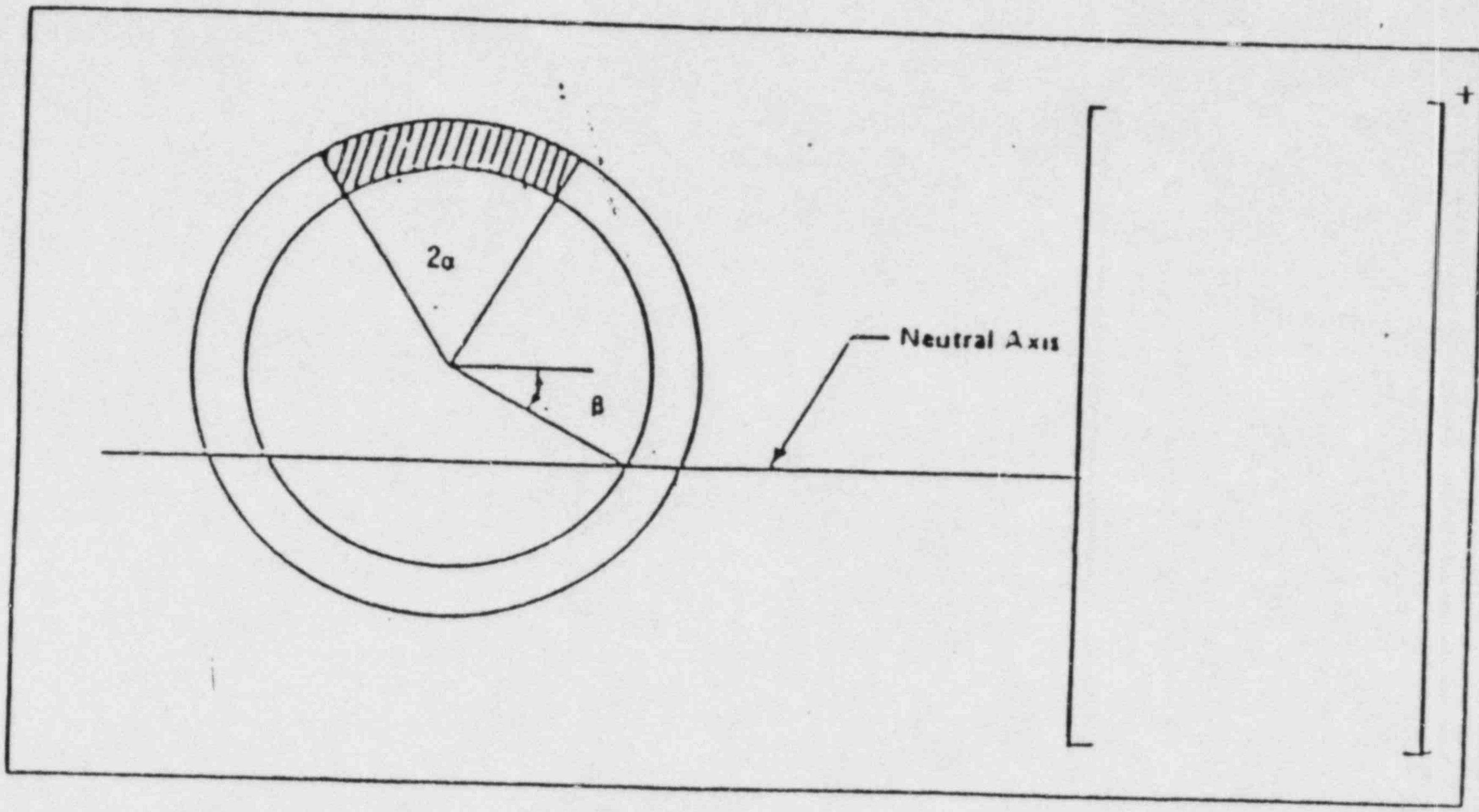
FIGURE 1 REACTOR COOLANT PIPE



[ ] +a,c,e

FIGURE 2 SCHEMATIC DIAGRAM OF PRIMARY LOOP SHOWING WELD LOCATIONS

Figure 3 [ ]+ a,c,e
Stress Distribution



]+ a,c,e

+a,c,e

FLAW GEOMETRY

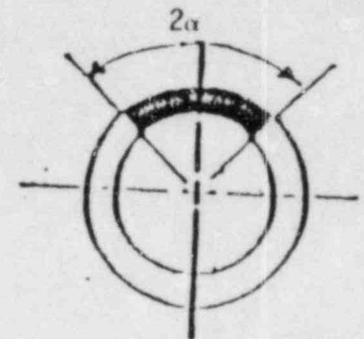


Figure 4 Critical Flaw Size Prediction

]a,c,e

FIGURE 5 TYPICAL CROSS SECTION OF [ ]a,c,e

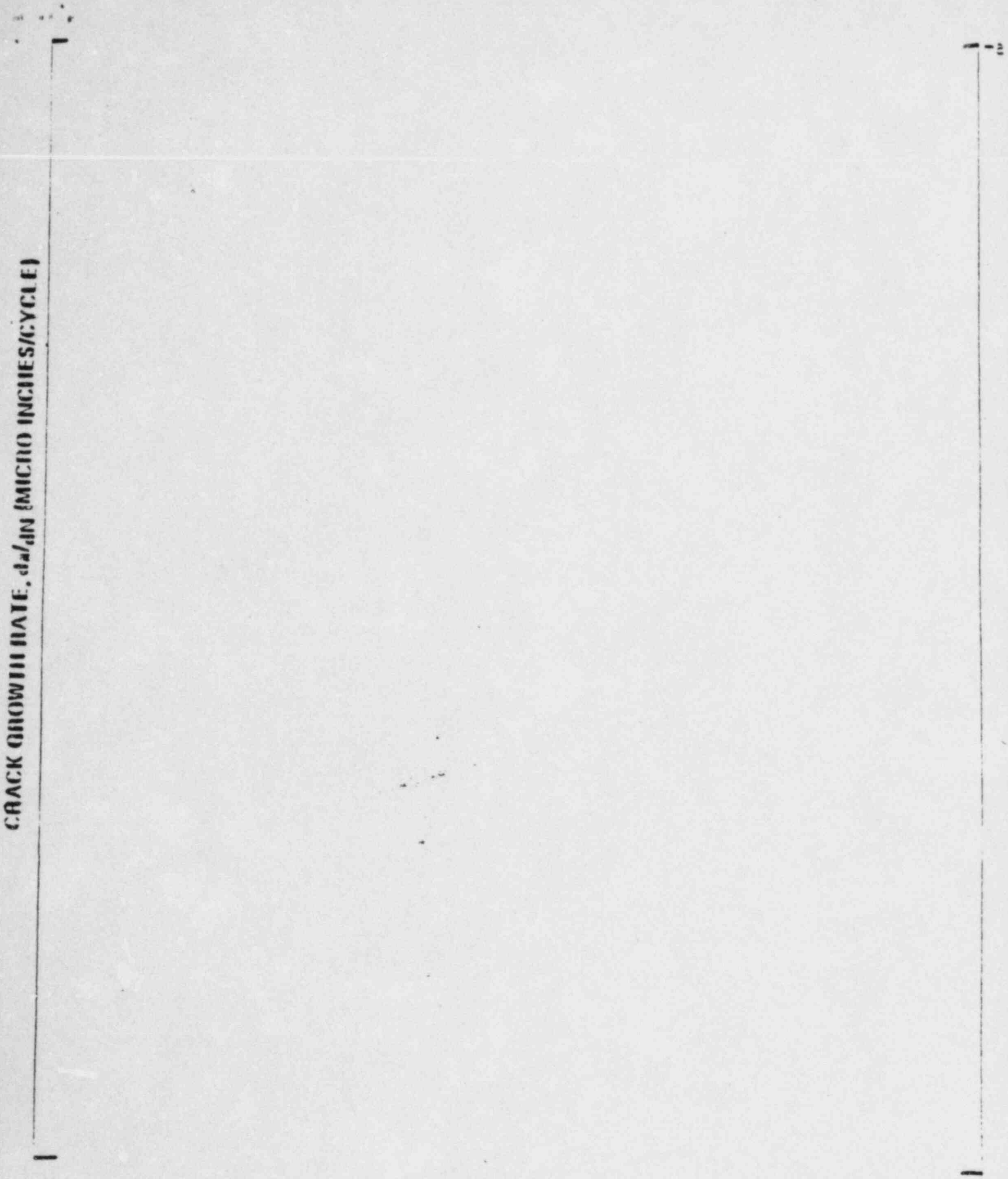


FIGURE 6 REFERENCE FATIGUE CRACK GROWTH CURVES FOR
[ ] +a,c,e

+a,c

Fig. 7 Reference Fatigue Crack Growth Law for Inconel 600 in a Water Environment at 600F.

a,c,e

Figure A-1 Pipe with A Through-wall Crack in Bending

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 corrections. Consequently, pipe whip and jet impingement effects of the postulated pipe break
MEB Q102 at these locations 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.

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 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) <u>High-Energy Piping</u>

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

1b) <u>Moderate-Energy Piping</u>

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 <u>Analytical Methods to Define Forcing Functions and Response Models</u>

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 (NRC), dated December 20, 1983, transmitting Westinghouse report justifying elimination of RCS primary loop pipe breaks fro 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.
Terminal ends should be considered to originate at a point adjacent to the required pipe whip restraints.	Penetration design is discussed in SAR Section 3.6.2.4. This section also discussed penetration guard pipe design criteria. 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

MEB
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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.

The crane wall is divided into two sections for analysis. Both the upper and
Q lower sections are modeled as space frames using STRUDL. For more details con-
220.44 cerning 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 (NRC), dated December 20, 1983, 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 I 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 <u>Fabrication</u>

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 <u>Materials</u>

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.

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

December 20, 1983

PROPRIETARY

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

WITHHELD FROM
PUBLIC DISCLOSURE

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

References: 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) Letter from H. B. Tucker (Duke Power Company) to H. R. Denton (NRC), dated November 18, 1983

Dear Mr. Denton:

References 1) and 3) 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. As a result of efforts by Westinghouse, the NRC, and Duke Power, we have concluded that it is technically feasible to eliminate these postulated pipe breaks. In addition, Westinghouse has assured Duke Power Company that the generic information previously submitted to the NRC to justify the elimination of RCS primary loop pipe breaks is applicable to the Catawba Nuclear Station.

As a result of the above developments, and in accordance with the statement in Reference 2) 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 letter is submitted. Duke Power hereby requests NRC approval for application of the "leak-before-break" concept to the Catawba Nuclear Station to eliminate postulated pipe breaks in the RCS primary loop from the plant structural design basis. A specific plant applicability report is included as Enclosure A to this letter. Because of the proprietary nature of this report, Enclosure A has been provided only to the addressee and Mr. James P. O'Reilly of the NRC. A non-proprietary version of the specific plant applicability report is included as Enclosure B and has been provided to others on the attached distribution list.

As Enclosure A contains information proprietary to Westinghouse Electric Corporation, it is supported by the attached letter (Attachment 1) and affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity

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the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. Correspondence with respect to the proprietary aspects of the Application for Withholding or the supporting Westinghouse affidavit should reference CAW-83-106, and should be addressed to R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Implementation of the leak-before-break concept 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.

Employment of the leak-before-break concept 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 and resultant flows from the leak-before-break analysis will be used when reactor cavity and subcompartment pressurization data are revised.

The impact on important design aspects of implementing leak-before-break on Catawba Nuclear Station 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. Duke Power has also evaluated the potential cost savings and operational benefits that result from the elimination of postulated pipe breaks in the RCS primary loop. A summary of the potential benefits which can be realized specifically from the elimination of these pipe breaks for Catawba Unit 2 is provided in Attachment 4. Note that these benefits total at least $2 million and involve an estimated 600 man-rem dose reduction over the life of Unit 2. Implementation of the leak-before-break concept will therefore be cost-effective as well as technically justifiable while resulting in improved overall plant safety.

Enclosure C consists of the revised Catawba FSAR pages associated with the elimination of RCS primary loop breaks, and it will be included in Revision 9 to the FSAR. This current request is for implementation on Unit 2 only; Duke Power will submit additional information prior to implementation on Unit 1.

Construction completion of the RCS primary loop pipe whip restraints at Catawba Nuclear Station Unit 2 is on hold pending an NRC ruling on this proposal. In order to realize the maximum advantage from the elimination of RCS primary loop ruptures, we request a decision by February 15, 1984.

If I can be of further assistance, or if a meeting with the Staff is deemed beneficial for a final resolution of this matter, please contact me.

Very truly yours,

H.B. Tucker / RSO

Hal B. Tucker

ROS/php

Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

NRC Resident Inspector
Catawba Nuclear Station

Mr. Robert Guild, Esq.
Attorney-at-Law
P. O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135½ Devine Street
Columbia, South Carolina 29205

Mr. Jesse L. Riley
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28207

ATTACHMENT 1

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

November 23, 1983
CAW-83-106

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

APPLICATION FOR WITHHOLDING PROPRIETARY

INFORMATION FROM PUBLIC DISCLOSURE

REFERENCE: Duke Power Company letter to NRC dated ~~November~~ December 1983

Dear Mr. Denton:

The proprietary material for which withholding is being requested in the reference letter by Duke Power Company is further identified in an affidavit signed by the owner of the proprietary information, Westinghouse Electric Corporation. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10CFR Section 2.790 of the Commission's regulations.

The proprietary material for which withholding is being requested is of the same technical type as that proprietary material previously submitted with application for withholding CAW-83-80.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Duke Power Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-83-106, and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/bek

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

Duke

~~8312300226~~

CAW-83-80

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared John D. McAdoo, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. D. McAdoo Jr.

John D. McAdoo, Assistant Manager
Nuclear Safety Department

Sworn to and subscribed before me this 26th day of September 1983.

Notary Public

NOTARY PUBLIC
ALLEGHENY COUNTY
1986

(1) I am Assistant Manager, Nuclear Safety Department, in the Nuclear Technology Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rule-making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hole certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Bases for the South Texas Project," dated September 1983, prepared by S. A. Swamy and J. J. McInerney.

The subject information could only be duplicated by competitors if they were to invest time and effort equivalent to that invested by Westinghouse provided they have the requisite talent and experience.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse because it would simplify design and evaluation tasks without requiring a commensurate investment of time and effort.

Further the deponent sayeth not.

ATTACHMENT 2

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

STRUCTURES, SYSTEMS, COMPONENTS, PROGRAMS 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 Detection Systems	No change
Environmental Qualification Program	No change

*Due to small hot gaps, the hot leg pipe whip restraints currently receive relatively small loadings from postulated main steam pipe breaks. It has been shown that the Steam Generator column supports are adequate to support the additional load in the absence of the hot leg pipe whip restraints. Also, an analysis is being performed to show that the reactor coolant loop loadings from the main steam pipe breaks will be acceptable without the hot leg pipe whip restraints.

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	Erection Status Catawba Unit 2
1. Reactor vessel inlet nozzle	1. Cold Leg Nozzle Break Restraint (wagon wheel)	1. Structure installed without shims
2. Reactor vessel outlet nozzle	2. Hot Leg Nozzle Break Restraint (wagon wheel)	2. Not installed
3. Steam generator inlet nozzle	3. Hot leg pipe whip restraint	3. Structure installed without shims
4. 50° elbow in the intrados (longitudinal slot)	4. Hot leg pipe whip restraint	4. Structure installed without shims
5. Steam generator outlet nozzle	5. Crossover leg pipe whip restraint (vertical run)	5. Structure installed with shims
	Crossover leg elbow restraints	Compression blocks installed without shimming
6. Reactor coolant pump inlet nozzle (pump suction)	6. Crossover leg elbow restraints	6. Compression block installed without shims
7. Crossover leg closure weld	7. Crossover leg elbow restraints	7. Compression blocks installed without shimming
8 Reactor coolant pump outlet	8. None	

ATTACHMENT 4

Estimated Cost Savings/Operational Benefits
for Elimination of Primary Loop
Pipe Breaks on Catawba
Unit 2

Category	Cost Savings (1983 rates)	Operational Benefit
1. Elimination of RCS pipe whip restraints	$0.6M - Pipe whip restraint installation cost*	-Substantial improvement in quality of ISI
	$1.3M - Occupational radiation exposure over Unit 2 life	-Substantial improvement in personnel access results in dose reduction of 600 man-rem
	- Simplifies plant design by elimination of potential interferences with piping, hangers, impulse tubing, etc.	-Improved access for operation and maintenance -Reduced RCS heat loss to containment at whip restraint locations.
	$0.1M - Eliminates additional hold points during initial heatup for verifying pipe-restraint clearances	-Reduced risk of unanticipated pipe restraint for thermal growth and seismic movement. -Improvement in overall plant safety (NUREG/CR-2136)
2. Simplification of analysis associated with dynamic effects and loading conditions.	- Pressurization loadings reduced on primary shield wall , crane wall, operating floor, and subcompartment analys s.	-Simplification of analyses involving loadings due to future plant modifications.
TOTAL	$2 Million	600 man-Rem

*Of a total of 20 restraints, four have not been installed. Shimming work has not been performed on any of the restraints.