PAGE REPLACEMENT GUIDE FOR

AMENDMENT 70

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 70 to Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 70 consists of new and replacement pages for the PSAR text and Responses to NRC Questions.

Vertical margin lines on the right hand side of the page are used to identify changes resulting from NRC Questions and margin lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 70 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

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AMENDMENT 70

QUESTION/RESPONSE SUPPLEMENT

This Question/Response Supplement contains an Amendment 70 tab sheet to be inserted following Qi page Amendment 69, July 1982. Page Qi Amendment 70 is to be inserted following the Amendment 70 tab sheet.

Following new Question/Response pages will be inserted in PSAR Volume 25 behind the appropriate numbered tabs located within that volume.

The parenthesis beside each Question/Response shown indicates the number of pages associated with each Question/Response.

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Insert tabed page 810 at the beginning of the QCS810 series question/responses.

1.6 MATERIAL INCORPORATED BY REFERENCE

1.6.1 Introduction

This section identifies technical reports incorporated by reference into the PSAR. Some of the technical reports cited were produced for the LMFBR program under the direction of the Energy Research and Development Administration (ERDA) and, therefore, contain the disclaimer notice as required by ERDA manual Appendix 3201, Part II-D. In support of the construction permit application for the Clinch River Breeder Reactor Plant, however, any such disclaimer notice should be considered to be deleted and therefore of no effect.

1.6.2 References

24 36 1. Deleted.

42 2. WARD-D-0185, "Clinch River Breeder Reactor Plant Integrity of Primary and Intermediate Heat Transport System Piping in Containment", September 1977.

3. WARD-D-0115, "Development and Application of a Cumulative Mechanical Damage Function for Fuel Pin Failure Analysis in LMFBR Systems", April 1976.

4. WARD-D-0005, "Demo Code" LMFBR Demonstration Plant Simulation Model, Rev. 4.

5. WARD-D-0090, "CRBRP Decay Power Analysis", January 1976.

36 6. Deleted.

7. AI Report No. 99-TI-413-039, "EVTM/CLEM Full Scale Test Analysis" R.G. Hanson, issued August 15, 1975.

8. AI Report No. 99-TI-413-042, "Subscale Emissivity Test Analysis (EVTM)", D. Vanevenhoven, issued October 17, 1975.

9. "Hypothetical Turbine Missile Data and Probability of Occurrence for 3600-RPM-23-Inch LSB Unit for Use with Liquid Metal Cooled Fast
42 16 Breeder Reactor", General Electric Co., August , 1977

10a. CRBRP-3, Volume 1, "Hypothetical Core Disruptive Accident Considerations in CRBRP; Energetics and Structural Margin Beyond the Design Basis", Rev. 4.

10b. CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP, Assessment of Thermal Margin Beyond the Design Base", Rev. 4.

11. WARD-D-0178, "CRBRP; Closure Head Capability for Structural Margin Beyond Design Basis Loading", Revision 3, November 1978.

12. WARD-D-0174, "CRBRP; Active Pump and Valve Operability Verification Plan", April 1977.

13. WARD-D-0165, "Requirements for Environmental Qualification of CRBRP Class 1E Equipment", Rev. 6.

14. WARD-D-0218, "Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident", October 1978.

15. CRBRP-GEFR-00103, "CRBRP; An Analysis of Hypothetical Core Disruptive Events in the Clinch River Breeder Reactor Plant", April 1978.

TABLE 2.3-30

DESIGN BASIS ACCIDENT χ/Q VALUES FOR THE
EXCLUSION AREA BOUNDARY (EAB) AND LOW POPULATION ZONE (LPZ) DISTANCES[1]
33 FT. WIND SPEED AND DIRECTION; 200-FT TO 33-FT DELTA T
PERMANENT TOWER DATA
FEBRUARY 17, 1977 - FEBRUARY 16, 1978

	EAB	LPZ					
	0-2 Hr	0-2 Hr	0-8 Hr	8-24 Hr	1-4 Day	4-30 Day	Annual
Design Accident /Q Value[2]:	1.1E-3	2.6E-4	1.2E-4	8.4E-5	3.7E-5	1.2E-5	2.8E-6
Maximum Sector 0.5 Percentile /Q Value:	1.1E-3	2.6E-4	1.2E-4	8.4E-5	3.7E-5	1.2E-5	2.8E-6
Overall Site 5th Percentile /Q Value:	8.7E-4	2.3E-4	1.1E-4	7.7E-5	3.5E-5	1.1E-5	2.8E-6

Sector Dependent 0.5 Percentile /Q Values

Exclusion Area Boundary (2 hr only)			Low Population Zone[4]						
Direction[5]	Value	Distance (m)	Direction[5]	0-2 Hr	0-8 Hr	8-24 Hr	1-4 Day	4-30 Day	Annual
N	3.5E-4	681	N	8.8E-5	3.9E-5	2.6E-5	1.1E-5	3.2E-6	7.0E-7
NNE	8.2E-4	671	NNE	1.2E-4	5.2E-5	3.4E-5	1.4E-5	3.7E-6	7.7E-7
NE	1.0E-3	671	NE	1.5E-4	6.5E-5	4.3E-5	1.7E-5	4.7E-6	9.7E-7
ENE	1.1E-3	671	ENE	1.9E-4	8.4E-5	5.6E-5	2.3E-5	6.5E-6	1.4E-6
E	9.1E-4	718	E	1.9E-4	8.9E-5	6.0E-5	2.6E-5	8.0E-6	1.9E-6
ESE	7.8E-4	832	ESE	2.1E-4	9.3E-5	6.1E-5	2.5E-5	7.0E-6	1.5E-6
SE	9.8E-4	832	SE	2.6E-4	1.2E-4	8.4E-5	3.7E-5	1.2E-5	2.8E-6
SSE	8.9E-4	870	SSE	2.3E-4	1.0E-4	7.1E-5	3.0E-5	8.8E-6	2.0E-6
S	1.7E-4	1966	S	8.9E-5	3.9E-5	2.5E-5	1.0E-5	2.8E-6	5.8E-7
SSW	2.9E-4	1134	SSW	9.1E-5	3.8E-5	2.5E-5	9.8E-6	2.5E-6	5.0E-7
SW	6.0E-4	832	SW	1.5E-4	6.4E-5	4.2E-5	1.7E-5	4.4E-6	8.9E-7
WSW	5.4E-4	839	WSW	1.4E-4	6.4E-5	4.3E-5	1.8E-5	5.3E-6	1.2E-6
W	6.2E-4	839	W	1.6E-4	7.4E-5	5.0E-5	2.2E-5	6.5E-6	1.5E-6
WNW	6.4E-4	882	WNW	1.7E-4	7.9E-5	5.4E-5	2.4E-5	7.2E-6	1.7E-6
NW	5.3E-4	1008	NW	1.4E-4	6.4E-5	4.3E-5	1.8E-5	5.3E-6	1.2E-6
NNW	6.9E-4	756	NNW	1.3E-4	5.6E-5	3.7E-5	1.5E-5	4.1E-6	8.5E-7

1. Computed according to R.G. 1.145
2. Greater of maximum sector 0.5 percentile value and 5th percentile overall
3. Computed according to R.G. 1.111
4. Distance of approximately 2.5 miles
5. Direction from which wind is blowing

TABLE 2.3-31

FIFTIETH PERCENTILE χ/Q VALUES FOR EAB AND LPZ DISTANCES[1]

33-FT WIND SPEED AND DIRECTION; 200-FT TO 33-FT DELTA T

PERMANENT TOWER DATA

FEBRUARY 17, 1977 - FEBRUARY 16, 1978

	Distance (miles)	50th Percentile χ/Q Values (sec/m^3)				
		0-2 Hr	0-8 Hr	8-24 Hr	1-4 Day	4-30 Day
Minimum Exclusion Area Boundary	0.42	1.01E-3	1.55E-4	1.23E-4	7.69E-5	9.06E-5
Low Population Zone Distance	2.5	1.59E-4	2.30E-5	3.58E-6	2.29E-6	2.60E-6

1. Values computed according to R.G. 1.4 (6/74). These maximum sector values occurred in the west-northwest or northwest sectors (east-southeast or southeast wind direction).

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3.7.2.1.2 Seismic Category I Systems and Components

The analysis of Seismic Category I systems and components is determined by a detailed dynamic analysis using either the response spectrum method or the time history method. The analysis is performed on a multi-mass mathematical representation of the system or components. A sufficient number of masses with their appropriate degrees of freedom are used in the model to adequately describe the behavior of the structural system, and to insure an accurate determination of the dynamic response. Significant non-linearities, such as gaps or clearances between PCRS components, are included in the mathematical model. In this case, a nonlinear time history analysis is performed, which considers the impact forces generated at the gap locations. Non-symmetrical features of geometry, mass, and stiffness, are modeled to include their torsional effects in the analysis. Descriptions of a preliminary reactor system linear model and a preliminary PCRS non-linear model are given in Section 3.7.3.15.

The methods of response spectra analysis and time history analysis are described in a number of publications. A description of these analyses techniques is provided in Appendix 3.7-A.

The system or component is analyzed with the seismic input (floor response spectra or time histories) derived at the particular points of support on the structure. All significant modes of the mathematical model are included in the analysis. The significant, dynamic response modes are those predominant modes which contribute to the total, combines modal response of the system. Other modes, whose inclusion in the square root of the sum of the squares modal summation have anegligible effect on the total response would not necessarily be used. With this procedure the number of modes included will be such that inclusion of additional modes will not result in more than a 10% increase in responses. Where the response spectrum method is used, the individual modal responses are combined by the square root of the sum of the squares, except for closely spaced modes (frequencies less than about 10% apart) where the modal responses are combined by the absolute sum. The analysis is performed independently in each of the two horizontal directions, and the vertical direction. Similar effects obtained for each of the three directions are combined by the square root of the sum of the squares. This is consistent with Regulatory Guide 1.92.

A simplified analysis based on a single mass model or an equivalent static load method may be used when it can be demonstrated that the simplified analysis provides adequate conservatism. For the simplified analysis, the equivalent static force, F_s, is distributed proportional to the mass of the component, and is calculated by the following equation:

$$F_s = 1.5 \ W \ A_s$$

where W is the total weight of the component, and A_s is the maximum peak acceleration of the response spectra, which apply at the points of support of the component. Components whose fundamental frequencies are greater than 33 Hz in any direction, are assumed to be rigid in that direction and may be designed for at least the maximum acceleration at their supports.

All systems and components under the jurisdiction of the ASME Section III Nuclear Power Plant Components Code will be designed to accommodate seismic loadings in combination with other loadings without producing total combined stresses in excess of those allowed by the Code. For elevated temperatures, applicable ASME-III Code Cases and RDT Standards will also apply. Stresses resulting from load combinations which include the OBE are limited to those given by the Code for Upset Conditions. These limits are intended to assure that the reactor will be able to continue or resume operation without undue risk to the health and safety of the public. For the load combinations which include the SSE, Faulted Condition allowables are generally applicable. The load combinations for the SSE and OBE are given in Appendix 3.7-A.

TABLE 3.7-7

LOCATION OF NODE POINTS

Node No. for Horizontal Directions	Corresponding Node No. for Vertical Direction	Distance from Center of Containment Building (Ft.)		Elevation (Ft.)	Applicable Nuclear Island Building
		x (E-W)	y (N-S)		
1	2	0.0	0.0	965.72	Reactor Containment
2	3	0.0	0.0	948.67	Reactor Containment
3	4	0.0	0.0	931.43	Reactor Containment
4	5	0.0	0.0	915.21	Reactor Containment
5	6	0.0	0.0	899.00	Reactor Containment
6	7	0.0	0.0	876.00	Reactor Containment
7	8	0.0	0.0	856.00	Reactor Containment
8	9	0.0	0.0	842.00	Reactor Containment
9	10	0.0	0.0	836.00	Reactor Containment
10	11	0.43	2.78	816.00	Reactor Containment
11	12	1.57	- 2.22	800.00	Reactor Containment
12	13	1.52	- 2.06	783.75	Reactor Containment
13	14	0.67	6.8	774.00	Reactor Containment
14	15	2.38	- 3.11	766.00	Reactor Containment
15	16	4.62	- 1.48	752.66	Reactor Containment
23	24	26.37	-161.64	884.00	Reactor Service
24	25	40.28	-157.55	869.00	Reactor Service
25	26	39.87	-157.86	857.00	Reactor Service
26	27	42.35	-170.03	840.00	Reactor Service
27	28	15.39	-160.41	816.00	Reactor Service
28	29	10.69	-160.57	797.00	Reactor Service
29	30	12.40	-159.35	779.00	Reactor Service

TABLE 3.7-7 (Continued)
LOCATION OF NODE POINTS (Cont.)

Node No. for Horizontal Directions	Corresponding Node No. for Vertical Direction	Distance from Center of Containment Building (Ft.)		Elevation (Ft.)	Applicable Nuclear Island Building
		x (E-W)	y (N-S)		
30	31	18.13	-156.74	765.00	Reactor Service
31	32	4.23	-150.01	755.00	Reactor Service
32	33	20.72	209.78	886.00	Steam Generator
33	34	-162.79	132.76	883.00	Steam Generator
34	35,60	- 26.68	155.13	873.00	Steam Generator
35	36	- 34.93	112.82	857.00	Steam Generator
36	37	-103.75	129.68	846.00	Steam Generator
37	38,61	- 47.77	127.34	837.00	Steam Generator
38	39	- 53.74	119.41	816.00	Steam Generator
39	40,62	- 13.01	148.39	806.00	Steam Generator
40	41	- 38.68	117.15	794.00	Steam Generator
41	42,63	- 21.94	131.55	787.00	Steam Generator
42	43,64	- 21.58	127.52	765.00	Steam Generator
43	44	- 37.29	139.32	746.00	Steam Generator
59 (N-S)	67	- 18.11	30.78	733.00	Common Base Mat
60 (E-W)	67	- 18.11	30.78	733.00	Common Base Mat

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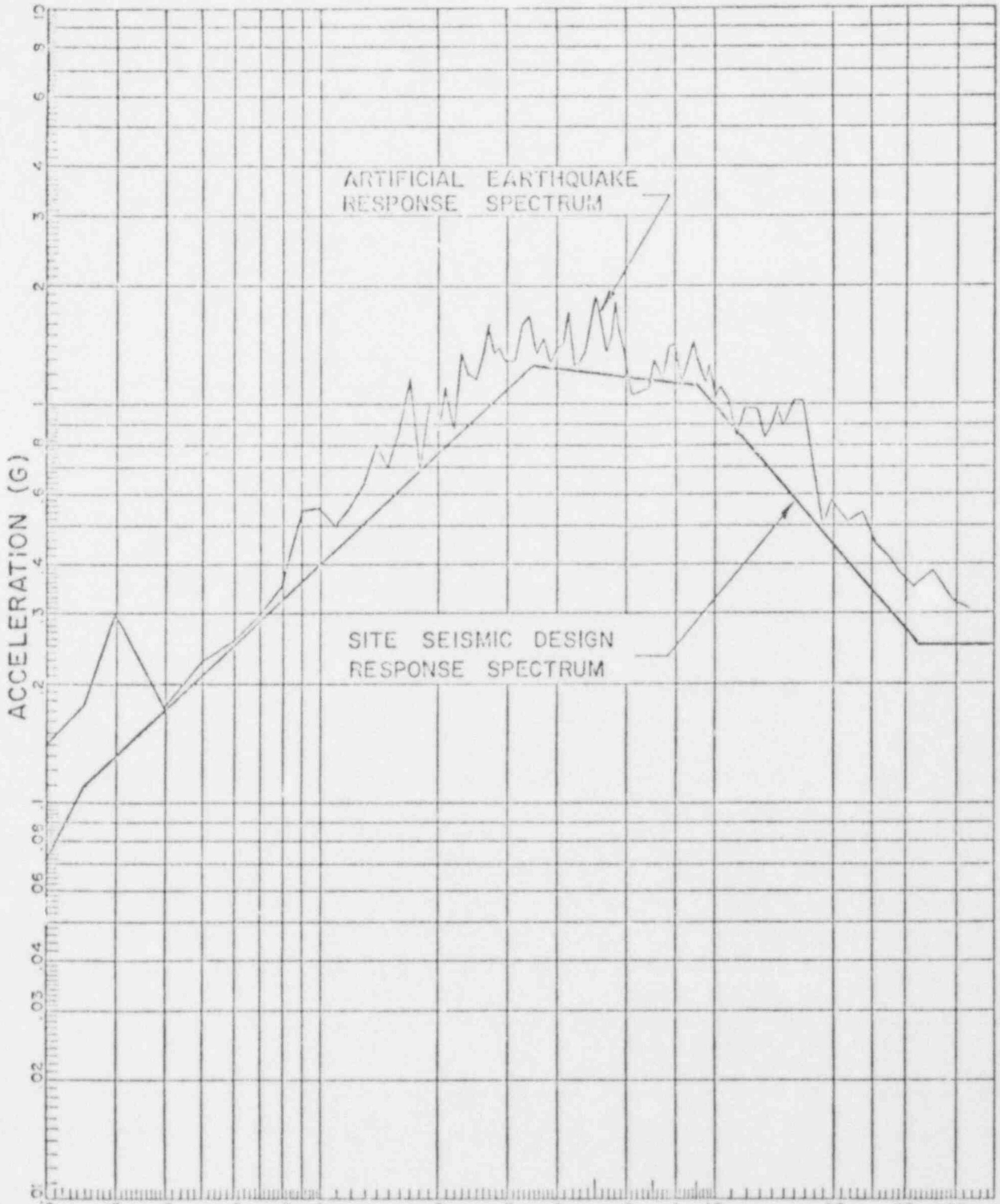
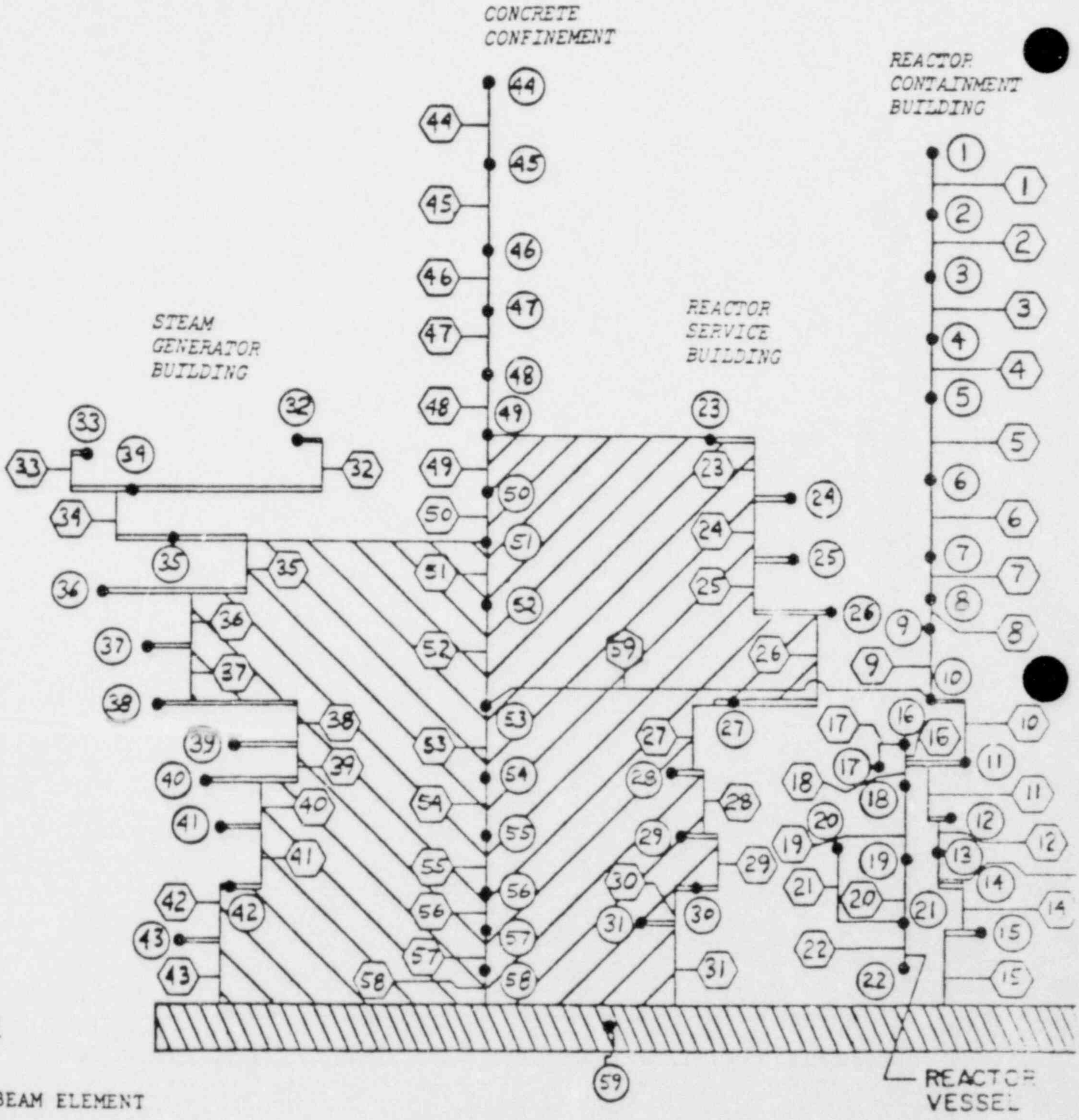


SAFE SHUTDOWN EARTHQUAKE
DESIGN RESPONSE SPECTRA
VERTICAL MOTION 1% DAMPING

Figure 3.7-15



LEGEND

BEAM ELEMENT

RIGID LINK

MASS POINT

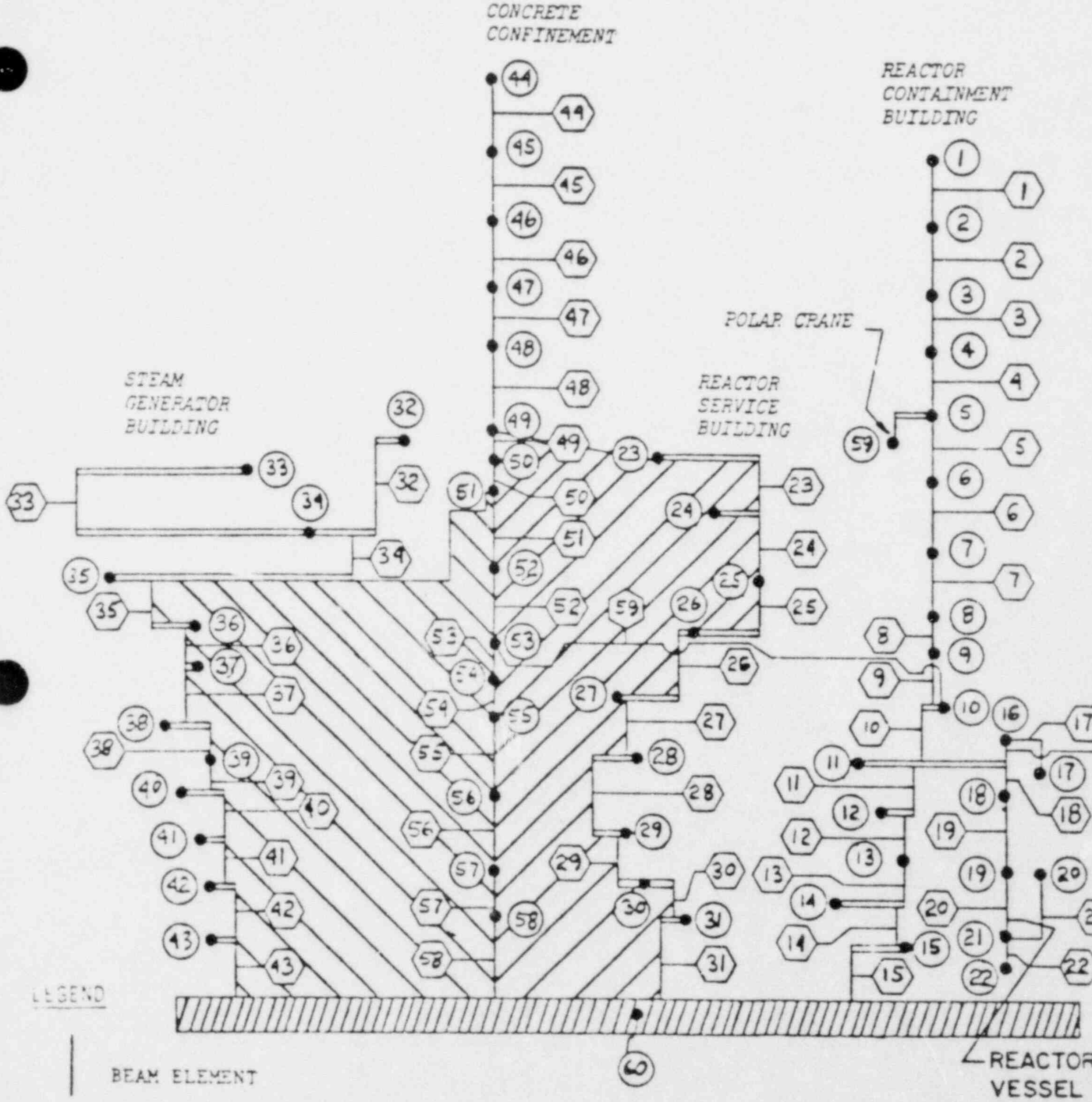
BEAM ELEMENT

FLEXIBLE TIES BETWEEN NODAL POINTS INCLUDING CROSS-COUPLING

* FOUNDATION SPRINGS NOT SHOWN

CRBRP BUILDING MATHEMATICAL MODEL FOR THE N-S EARTHQUAKE

FIGURE 3.7-16



LEGEND

BEAM ELEMENT

RIGID LINK

MASS POINT

BEAM ELEMENT

FLEXIBLE TIES BETWEEN NODAL POINTS INCLUDING CROSS-COUPLING

* FOUNDATION SPRINGS NOT SHOWN

CRBRP BUILDING MATHEMATICAL MODEL FOR THE E-W EARTHQUAKE

FIGURE 3.7-16a

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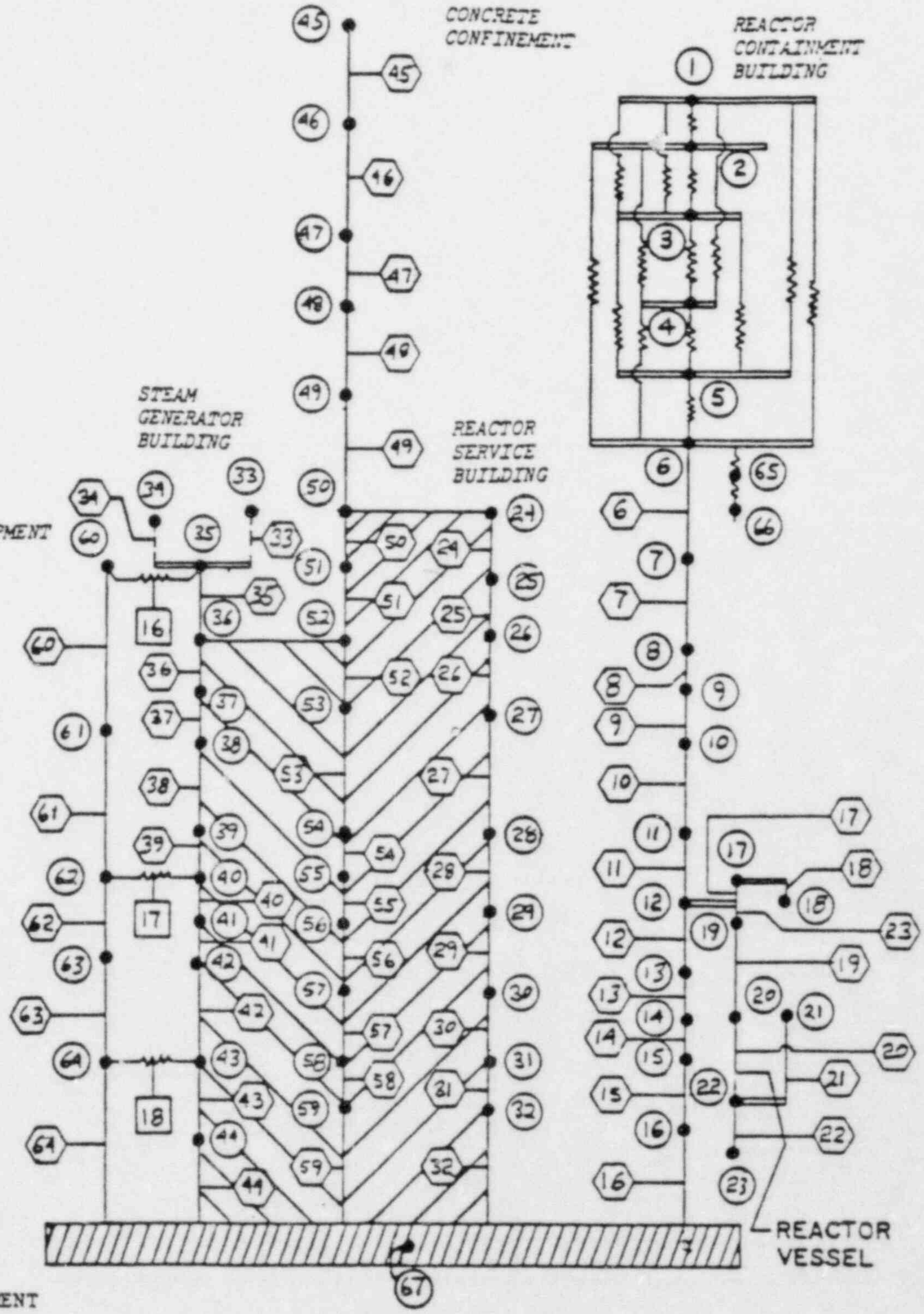

LEGEND

BEAM ELEMENT

RIGID LINK

MASS POINT

BEAM ELEMENT

FLEXIBLE TIES BETWEEN NODAL POINTS INCLUDING CROSS-COUPLING

* FOUNDATION SPRINGS NOT SHOWN

SPRING ELEMENT

CRBRP BUILDING MATHEMATICAL MODEL FOR THE VERTICAL EARTHQUAKE

FIGURE 3.7-16b

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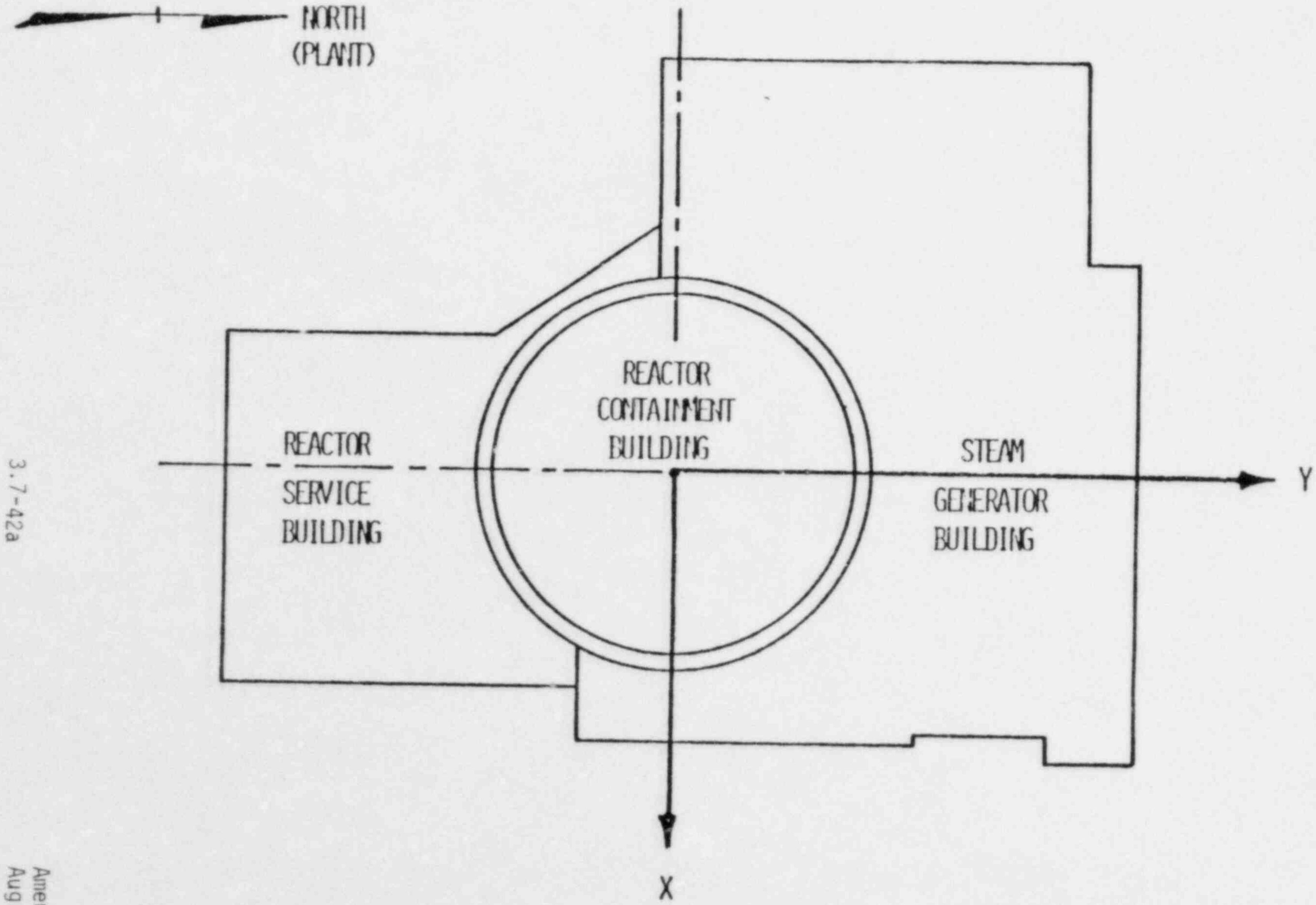

FIGURE 3.7-16c CRBRP BUILDING NODES COORDINATE SYSTEM - PLAN

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9976-2

3.7-42b

Amend. 46
Aug. 1978

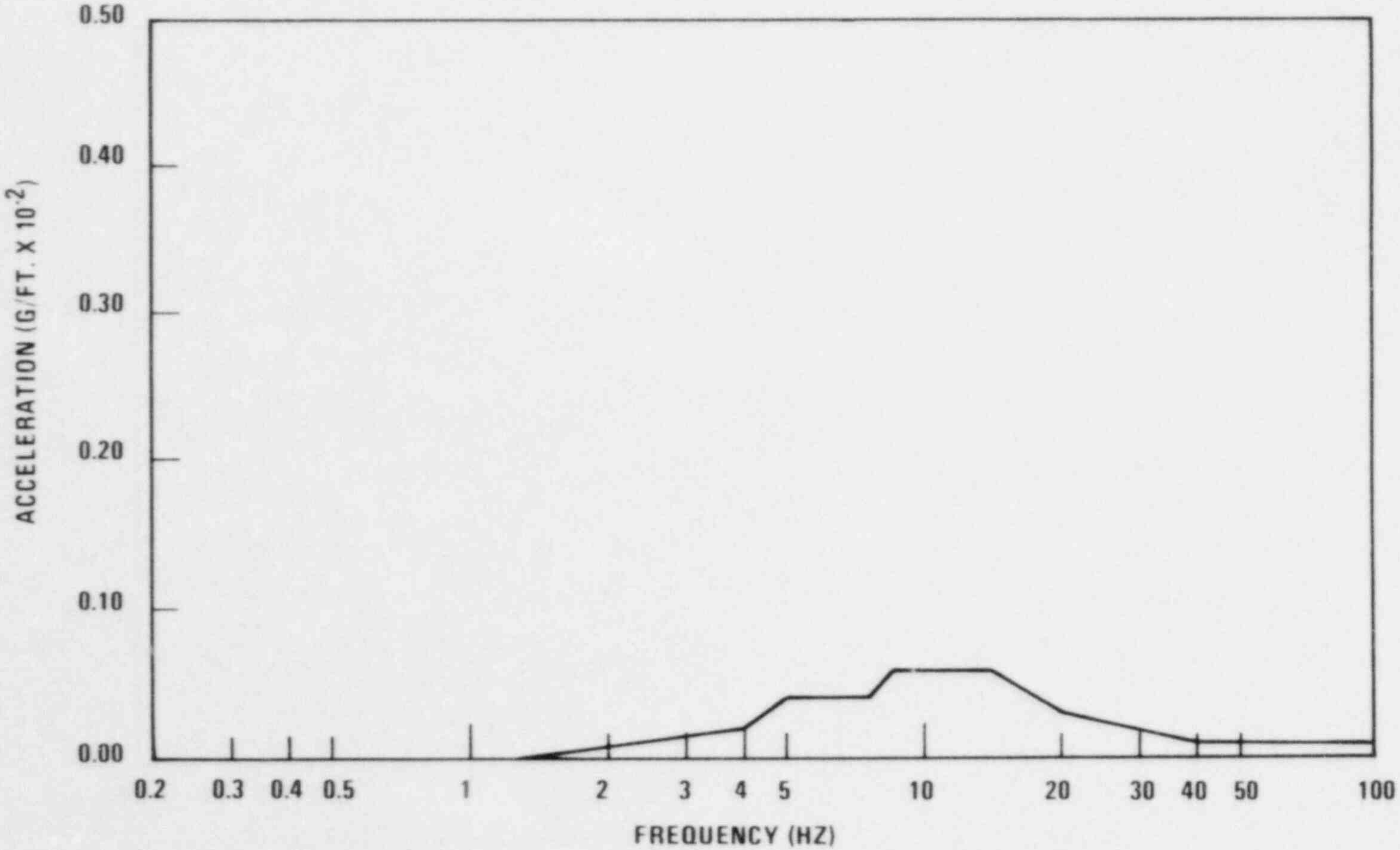

Figure 3.7-16D 0.250g SSE Torsional E-W Response Spectrum for Common Base Mat at El. 733' Node 48 (3% Damping)

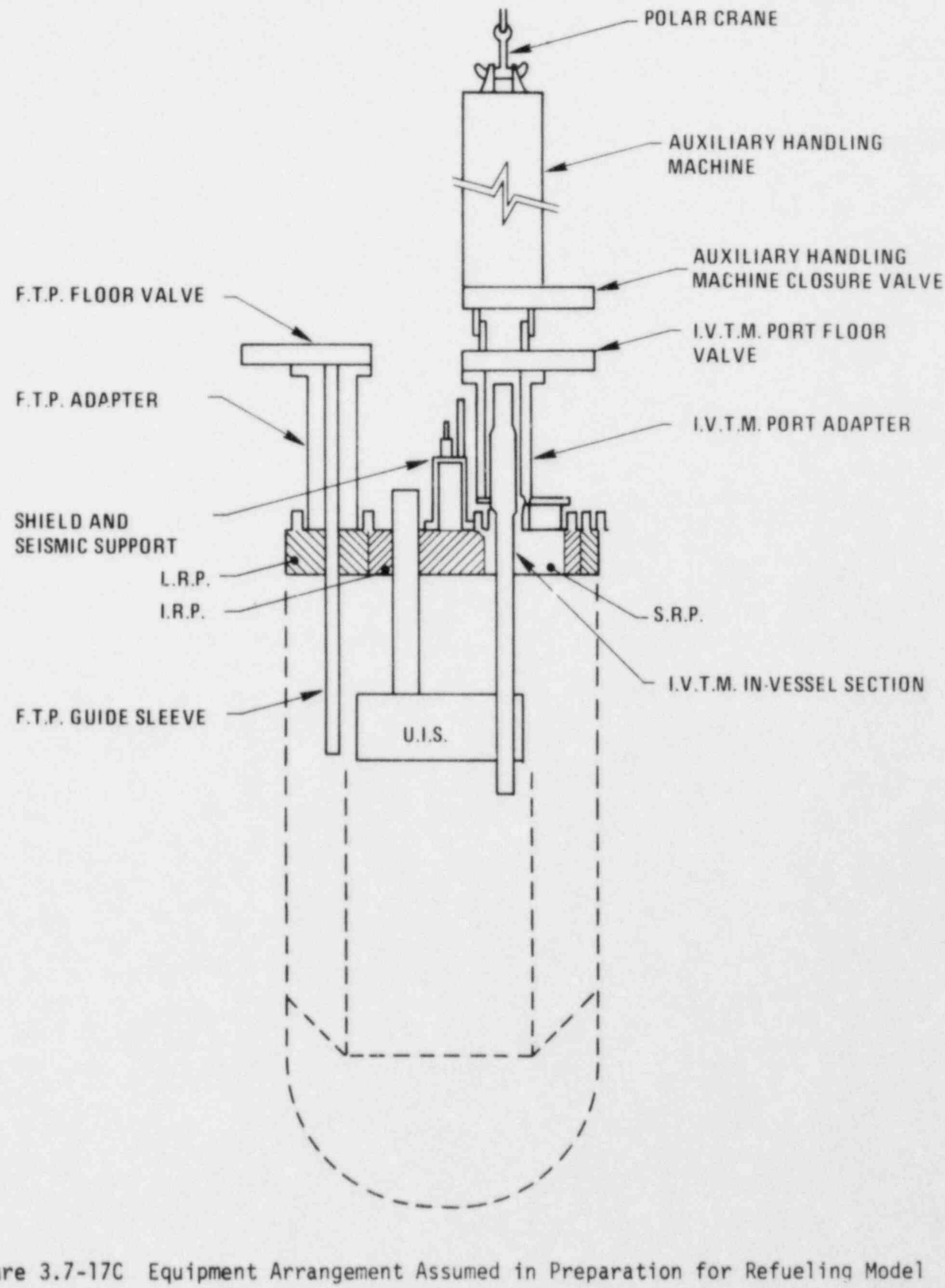

Figure 3.7-17C Equipment Arrangement Assumed in Preparation for Refueling Model

9859-3

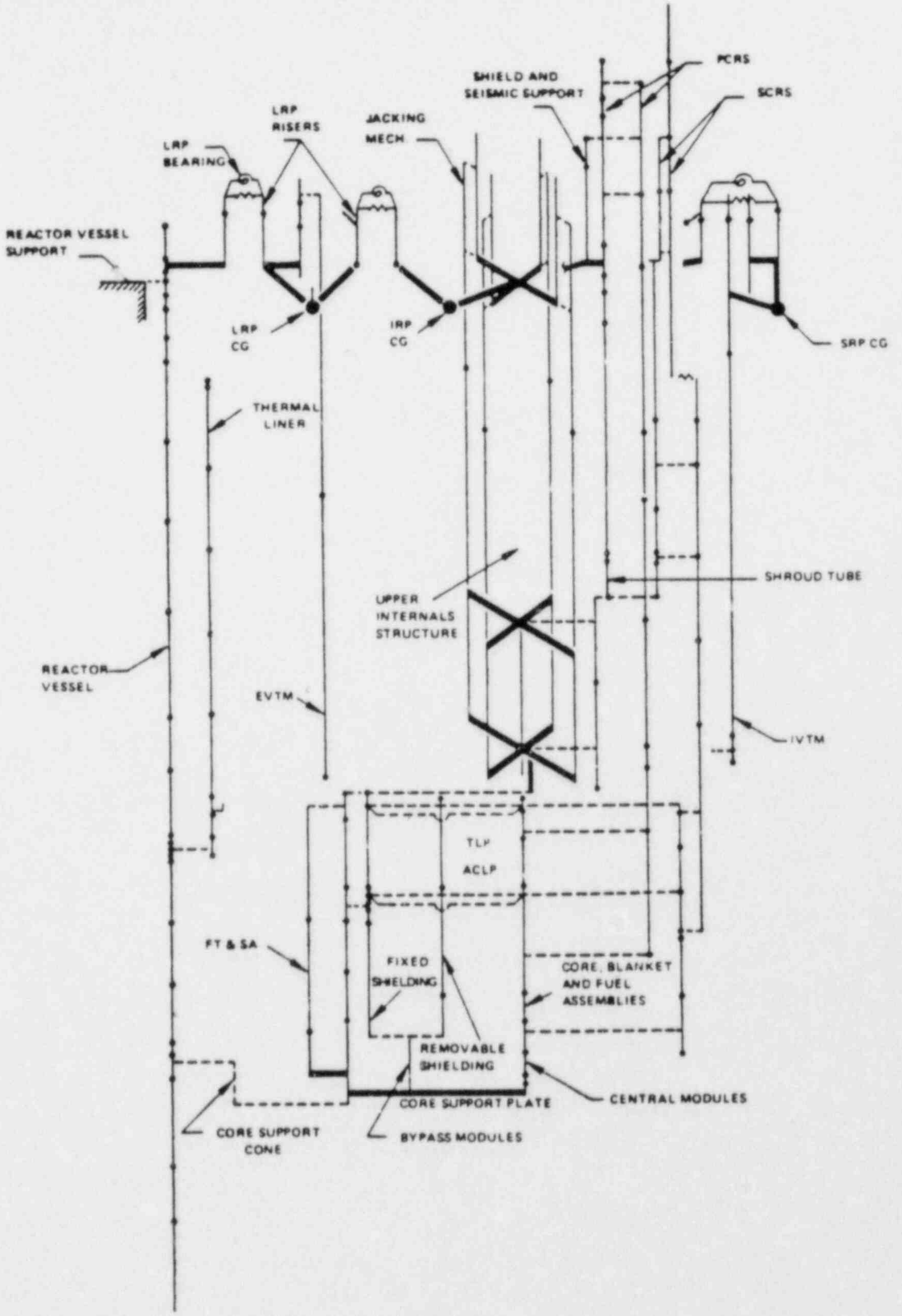

Figure 3.7-17D Schematic of Reactor System Finite Element Model

9859-5

6.1 Dynamic Analyses

Seismic Category I and II structures, systems, and components shall be analyzed by a detailed dynamic analysis using either time history methods or the response spectrum method. Other methods of dynamic analysis which provide an acceptable solution may be used but the justifications and procedures shall be submitted to W-LRM for review and approval. A simplified analysis such as that based on an equivalent static load method may be used if it can be demonstrated that the simplified method provides adequate conservatism.

Analytical procedures of the detailed dynamic analysis and of a simplified analysis are given in Attachment A.

The analysis will include the effects of the dynamic coupling between major components of the system and the soil-structure interaction effects. A sufficient number of modes of the mathematical model which represents the structural system shall be included in the analysis to assure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10% increase in responses. Where the response spectrum method is used, the individual modal responses shall be combined by the square root of the sum of the squares, except for closely spaced modes (frequencies 10% or less apart) where the modal responses shall be combined by the absolute sum (see Attachment A). Supports of structural systems (such as piping) may be subjected to different accelerations; i.e., different response spectra for the particular support location. In this case, the different response spectra should be super imposed to yield an envelope response spectrum to be used in the response spectrum analysis.

The system will also be analyzed to determine the effects of the relative displacements at their supports.

The relative displacements should be imposed on the mathematical model as separate static displacements unless other conservative methods of analysis are employed or a time history analysis is performed. In the latter case, the appropriate time histories are used at each anchor point location so that the differential effects are inherently included in the analysis. The relative displacements are imposed in the most unfavorable manner to satisfy input motions which may be out of phase from each other. The effects resulting from the separate analysis for relative displacements should be combined absolutely with those resulting from the seismic-induced inertial loadings.

Time history analyses of the supporting structure are required to determine the response motin at the location on the structure of the supported component. This motion is used as input to the analysis of the supported component and may be in the form of motion time history or floor response spectra.

The floor response spectra to be used for design of components (subsystems) are calculated at frequency intervals not greater than those given in Table 3.7A-3. These spectra are widened in frequency to account for the effect of possible frequency variation of the structure, and smoothed (in accordance with Regulatory Guide 1.122). Since floor response spectra for both upper and lower bounds of soil moduli are produced by the buildings anaylsis, a widening of ± 10% prior to smoothing is considered adequate. In addition, the spectra are enveloped for the upper and lower bounds of soil moduli to obtain design response spectra. When time history analysis of subsystems are used directly to obtain loadings or effects (such as displacements, inertia forces, stresses) the effect of possible frequency variation of the structure shall also be considered. An acceptable method using the same time history data with three different time intervals is given in NRC standard Review Plan 3.7.2. Other methods may be used if justified, such as development of synthesized (design) time
46 histories whose response spectra envelop the design response spectra.

Seismic Category III structures, systems and components may be
6 analyzed using equivalent static loadings. The equivalent static loadings for structures should satisfy the requirements of the Standard Building Code, Zone 2. For the analysis of systems and components, when required by local codes and the Standard Building Code, the determination of the equivalent static loadings should consider the amplification of the ground motion at the support point of the component on the structure. The simplified analysis method of Attachment A may be used as an option for this analysis, although it is not required.

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6.2 Mathematical Modeling

Structures and components of physically connected systems may by idealized as a combined single mathematical model, or subdivided into smaller

ATTACHMENT A

ANALYTICAL PROCEDURES

A.1 Detailed Dynamic Analyses

Two of the methods used to perform a suitable, detailed dynamic analysis of a structural system are the time history analysis and the response spectrum analysis. The equations of motion are solved either by direct integration or modal superposition. Using modal superposition th normal modes of the system are obtained along with the natural frequencies, mode shapes and participation factors. For the time history analysis, the forcing function consists of the earthquake motion as a function of time. For the response spectrum analysis, the input excitation is provided in the form of response spectra which give the spectral response motions (displacement, velocity, acceleration) as a function of the natural frequencies for the appropriate damping values.

A.1.1 Equations of Motion

The equations of motion of a multi-degree-of-freedom discrete-mass damped system subjected to an arbitrary support motion $\ddot{Y}_s(t)$, can be written as:

$$M\ddot{X}(t) + C\dot{X}(t) + KX(t) = -MI\ddot{Y}_s(t) \qquad (1)$$

where:

M = Mass matrix

C = Damping matrix which can be expressed as a linear combination of the mass and stiffness matrices

K = Stiffness matrix

I = Unit vector in direction parallel to support motion

$\ddot{Y}_s(t)$ = Time dependent support acceleration

$X(t)$, $\dot{X}(t)$ and $\ddot{X}(t)$ = Time dependent displacement, velocity and acceleration vectors, respectively.

In the direct integration approach, the coupled equations of motion are solved directly. In the modal analysis, using the orthogonality relations and expressing the displacements, velocities and accelerations in normal coordinates; i.e., $X(t) = \phi A(t)$, $\dot{X}(t) = \phi \dot{A}(t)$, and $\ddot{X}(t) = \phi \ddot{A}(t)$, the above coupled equations of motion (Eq. 1) may then be rewritten as the following uncoupled, normal equations of motion:

$$M_r\ddot{A}_r(t) + 2M_r\omega_r\xi_r\dot{A}_r(t) + K_rA_r(t) = -M_r\Gamma_r\ddot{Y}_s(t)$$

where:

$M_r = \phi_r^T M \phi_r$ = Generalized mass for mode r.

$\xi_r = \dfrac{\phi_r^T C \phi_r}{2\omega_r \phi_r^T M \phi_r}$ = Generalized damping factor for mode r.

$K_r = \phi_r^T K \phi_r$ = Generalized stiffness for mode r.

$\Gamma_r = \dfrac{\phi_r^T M I}{\phi_r^T M \phi_r}$ = Participation factor for mode r.

ω_r = Undamped circular frequency of mode r.

ϕ_r = Mode shape vector for mode r.

The undamped circular frequencies, ω, may be calculated from:

$$\left| K - \omega^2 M \right| = 0 \qquad (3)$$

and the mode shape vector for mode r can be obtained from:

$$[K - \omega_r^2 M]\ \phi_r = 0 \qquad (4)$$

The solution of the differential equation (Eq. 2) for the case of at-rest initial conditions is:

$$A_r(t) = \frac{-\Gamma_r}{\omega_r \sqrt{1-\xi_r^2}} \int_0^t \ddot{Y}_s(\tau) e^{-\xi_r \omega_r (t-\tau)} \sin\left[\omega_r \sqrt{1-\xi_r^2}(t-\tau)\right] d\tau \qquad (5)$$

For small damping factors, ξ_r, the above solution may be approximated by:

$$A_r(t) = \frac{-\Gamma_r}{\omega_r} \int_0^t \ddot{Y}_s(\tau) e^{-\xi_r \omega_r (t-\tau)} \sin\left[\omega_r (t-\tau)\right] d\tau \qquad (6)$$

For closely spaced modes, that is, two consecutive modes whose frequencies differ from each other by 10% or less of the lower frequency, the combination is performed as follows:

1. Divide closely spaced modes into groups which include all modes with frequencies lying between the lowest frequency in the group and 10% higher.

2. Take the absolute sum of the modes in each group.

3. Take the square root of the sum of the squares of the absolute sums (obtained in Step 2) and of the remaining modes which are not closely spaced.

For example, assume the total number of n modes can be divided into two groups of closely spaced modes, group A and group B, and modes j thru n which are not closely spaced. Further, assume that group A contains modes 1, 2 and 3; and group B contains modes 4, 5, and 6. The total combined displacement X_{ic}, would then be:

$$X_{ic} = \left[(|X_{i1}| + |X_{i2}| + |X_{i3}|)^2 + (|X_{i4}| + |X_{i5}| + |X_{i6}|)^2 + X_{ij}^2 + \ldots\ldots\ldots + X_{in}^2 \right]^{1/2} \qquad (18)$$

Other methods such as the "Ten Percent Method" and the "Double Sum Method" given in Regulatory Guide 1.92 are acceptable.

The above combination is performed as a last step (for a particular
46 earthquake direction) for the same type of effects individually, that is, either displacements, accelerations, moments, etc.

A.2 Simplified Analysis

Where it can be demonstrated that the use of a simplified analysis provides adequate conservatism, any structural system (1) for which the motion of its supports is known and represented by response spectra; and (2) which is of a regular nature; i.e., does not have large discontinuities in mass and is not of a highly irregular configuration, can be analyzed by the following approximate method:

Regardless of the natural frequencies of the structural system, the equivalent static force may be taken as:

$$F_s = 1.5\ W\ A_s \qquad (19)$$

where:

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F_s = Equivalent static force distributed proportional to the mass of the system.

W = Weight of the system including liquid contents.

A_s = Maximum peak acceleration of the response spectra which apply at the points of support of the structural system. (In "g" units)

A factor less than 1.5 in Equation 19 may be used if adequate justification is provided.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment (Not Applicable)

3.8.2 Steel Containment System

3.8.2.1 Description of the Containment

The Containment Vessel is a low leakage, free standing, all welded steel vessel with a steel lined reinforced concrete bottom. The steel vessel consists of a vertical cylinder with an ellipsoidal-spherical top dome.

The cylindrical part of the vessel has an inside diameter of 186 feet and has a height of approximately 169 feet from the flat bottom liner at the base to the spring line of the dome. Figure 3.8-13 presents the details of the vessel. The containment has four ring stiffeners, all above the operating floor. The top two stiffeners are connected by 60 equally spaced radial vertical gussets and support the polar crane. Figure 3.8-14 presents details of the polar crane girder support.

The cylindrical steel shell is embedded in concrete from the foundation up to the elevation of the operating floor. On the inside of the Containment Vessel, there is a continuous reinforced concrete wall comprising the peripheral boundary of the internal concrete structure. Butting against the outside face of the steel shell from elevation 733 feet up to the elevation of the underside of the operating floor, there is a 22" thick reinforced concrete wall designed to prevent buckling of the steel shell. Neither of the peripheral walls are conected to the Containment Vessel except for shear keys which are provided at the operating floor level to transfer horizontal seismic loads. The shear keys are designed to allow free axial growth of the cylindrical shell. The embedded steel shell will be coated with a bond breaker prior to concrete placement to prevent bond between the vessel and the concrete walls. The walls are not considered part of the Containment.

The bottom liner is anchored to a reinforced concrete foundation mat, 15 feet thick. On top of the liner, a reinforced concrete fill slab provides protection for the liner. The Containment Vessel is anchored to the foundation mat by means of an embedment skirt.

The foundation mat of the Containment Vessel is part of the common foundation mat of the Nuclear Island Structures. The limits of the ASME Section III, Division 2 Code boundaries for the foundation mat are shown on Figure 3.8-13.

Alumina-silica insulation is attached to the inside surface of the Containment Vessel from Elevation 816 feet to Elevation 825 feet to limit the Containment Vessel thermal stresses at the operating floor elevation during the Design Basis Accident. A minimum of 6 inches of insulation, having a value of 0.035 BTU/hr-ft-^{o}F, is required to limit the shell temperature at elevation 816', to 130^{o}F.

The vessel includes: its shell, one horizontal gland girder, two horizontal stiffeners, a 1/4" bottom liner plate, one access airlock, one emergency egress airlock, vacuum relief system, one equipment hatch, penetrations, inspection ladders, miscellaneous appurtenances and attachments. The equipment hatch has a inside diameter of 44'-6" to enable passage of large equipment and components into the containment. To the west of the equipment hatch there is an equipment/personnel airlock for routine access to the RCB. To prevent buckling in the equipment hatch area the plate is stiffened by a frame structure. Details are shown in Figure 3.8-15. The configuration of the shell is shown in Figure 3.8-3. The design lifetime of the containment vessel is 30 years.

3.8.2.2 Applicable Codes, Standards and Specifications

3.8.2.2.1 Codes

The Containment Vessel will be designed, material procured, fabricated, installed and tested in accordance with the requirements of the ASME B&PV Code, Section III, Division 1, 1974 Edition with Addenda through Winter 1974 and Code cases 1713, 1714, 1809, 1682 and 1785 and ASME-III, Division 2, 1975 Edition, Subsection CC, for the steel lined concrete containment bottom and foundation mat concrete. The design shall also meet the requirements of the Class MC Section of RDT Standard E15-2T, "Requirements for Nuclear Components".

3.8.2.2.2 Design Specification Summary and Design Criteria

The Containment Vessel, including all access openings and penetrations will be designed such that the leakage of radioactive materials from the Containment under conditions of temperature and pressure resulting from the extremely unlikely faults could not cause undue risk to the health and safety of the public and will not result in potential offsite exposures in excess of guideline values of 10CFR100.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 Design Loads

The following loads shall be used in the design of the Containment Vessel and Appurtenances.

D - Dead Load, including the weight of the steel containment vessel, penetration sleeves, equipment and personnel access hatches, and other attachments supported by the vessel, plus loads due to concrete shrinkage.

L - Live Loads, as applicable, including:

1. Penetration Loads (including seismic), as applicable
2. Floor Loads - 100 PSF
3. Walkways -200 lbs per linear foot
4. Equipment and Personnel Airlock Floor Load - 300 PSF or 40,000 lbs moving concentrated load
5. Emergency Airlock Floor Load -200 PSF or 10,000 lbs.
6. Polar Crane Loads
7. Construction Loads*
8. Painters Line Anchor - 2,000 lb. in any horizontal direction
9. Interior Scaffold -2,000 lb. each on any 2 adjacent clips Support Clips - combined with a Dead Load on all clips of 200 lbs. each.

W - Wind loads of 80 mph (ANSI A58.1-1972)

P_i - Internal Design Pressure (or Transient Pressure Loads)

P_e - External Design Pressure

P_t - Testing Pressure

T_o - Thermal loads due to temperature gradient through walls under normal operating conditions.

T' - Thermal loads due to temperature gradient through walls from accidents, such as major sodium fires.

T_t - Thermal load under testing temperature conditions.

E - Loads resulting from an Operating Basis Earthquake (OBE)

E' - Loads resulting from a Safe Shutdown Earthquake (SSE)

* A concrete placement load, resulting from using the vessel shell below operating floor elevation as the formwork for placing the reinforced concrete walls, and loads that are imposed by concrete forms when constructing the confinement shell. A snow load will be considered also during the construction period.

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R - Accident loads due to Extremely Unlikely Faults (such as interfacing loads from inner cell concrete structures).

Note (1)

The crane live load shall include, as appropriate, the vertical impact load and the lateral thrust load as determined in accordance with Reference 3.

Design Pressures and Temperatures

The design pressures and the associated design temperatures shall be as specified below:

Internal Design Pressure	10 psig
External Design Pressure	0.5 psig
Design Temperature	250°F 30

The design of the containment vessel may also consider a transient design pressure and attendant temperature loading due to extremely unlikely faults. Details of this information will be provided in the FSAR.

The operating condition containment atmosphere temperature and pressure are as follows:

Operating Condition Temperature = 70°F
Operating Condition Pressure = 0.0 psig
Lowest Service Metal Temperature = 15°F

3.8.2.3.2 Loading Combinations

The loading combinations for which the vessel and its appurtenances are to be designed shall be, but not limited to, those specified in Table 3.8-1. The containment design requirements and limits shall be in accordance with ASME-III, Article NE-3000.

For conditions where seismic loads are involved, the design analysis requirements and criteria as contained in Section 3.7 shall also be met.

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3.8.3.1.2 Head Access Area (HAA)

The head access area is located below the operating floor level and above the reactor cavity. The access area is of a square shape 44 feet long on each side and 14 feet high above the reactor head. The head access area is a reinforced concrete structure. Steel framing will be provided in this area to support the EVTM operations.

3.8.3.1.3 Primary Heat Transport System (PHTS) Cell

Each PHTS cell is a step type rectangular reinforced concrete structure. At its widest section, the cell is approximately 48 feet wide by 72 feet long. The cell is approximately 58 feet deep in the area housing the primary sodium pump and intermediate heat exchanger. The interior surfaces of the cells are provided with cell liners designed to contain hot sodium spills. The cells are designed to withstand accident pressure and temperature conditions as noted in Table 3.8-2.

3.8.3.1.4 Reactor Overflow Vessel and Primary Sodium Storage Vessel Cell

This cell is a step type rectangular reinforced concrete structure. The cell is approximately 26 feet wide by 69 feet long with 62 feet height at its deepest section. The interior surface of the cell is lined with carbon steel plate similar to the PHTS cells. The cell is designed to withstand accident pressure and temperature conditions noted in Table 3.8-2.

3.8.3.1.5 Other Cells

These cells are reinforced concrete structures with various sizes. The cells required to maintain a nitrogen atmosphere during operations will be lined and designed to the requirements noted in Table 3.8-2. Cell liners are described in Section 3A.8.

3.8.3.1.6 Fill Slab

A structure fill slab of suitable thickness will be provided over the bottom containment liner plate.

3.8.3.2 Applicable Codes, Standards and Specifications

3.8.3.2.1 Design Codes

Applicable provisions both mandatory and recommended of the following codes will be used in the design of the internal structures:

a. Code Requirements for Nuclear Safety-Related Concrete Structures (ACI-349).

b. Specification f the Design, Fabrication and Erection of Structural Steel for Buildings, ..erican Institute of Steel Construction, February, 1969, including Supplement 1 (11/70), Supplement 2 (12/71) and Supplement 3 (10/75).

c. Code for Concrete Reactor Vessels and Containments of ACI ASME Committee (ASME B&PV Code, Section III, Division 2 1975).

d. Applicable State Codes.

ACI-349 will be extensively used for the design of the internal structures. ACI-349 is generally based upon ultimate load design. Since loading combinations for the internal structures require that ultimate capacity of a section be always greater or equal to the imposed load combination, this code is best appropriate for the design of the above structures. Chapter 18 of ACI-349 will not be invoked since this is not relevant to the structures under discussion. For thermal stresses due to temperature gradient, the requirements of ACI-349 (Appendix A) will be used.

AISC specification will be applied to the structural steel members such as steel embedments, beams, equipment and pipe supports and restraint structures.

3.8.3.2.2 Structural Specifications

See subsection 3.8.4.3 for the appropriate structural specifications.

3.8.3.2.3 NRC Regulatory Guides

The design will meet requirements or basic intent of the following NRC Regulatory Guides:

a. 1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1/73)

b. 1.15 Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1, 12-28-72)

c. 1.29 Seismic Design Classification (Revision 1, 8/73)

d. 1.55 Concrete Placement in Category 1 Structures (6/73)

e. 1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants (Revision 1, December, 1973)

f. 1.61 Damping Values for Seismic Design of Nuclear Power Plants. (Oct. 1973)

g. 1.69 Concrete Radiation Shield for Nuclear Power Plants (Dec. 1973)

h. 1.92 Combine Modal Responses and Spatial Components in Seismic Response Analysis. (Rev.1 Feb. 1976)

i. 1.142 Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessel and Containment). (Rev. 1 Oct. 1981)

j. NUREG - 0554, Single-Failure-Proof Cranes for Nuclear Power Plants (May 1979)

3.8.3.2.4 ASTM Standards

All ASTM Standards to the extent they are referenced in the codes and standards noted in Section 3.8.3.2 and further specifically identified in other parts of Section 3.8.8, will be applied to the design of the facility.

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 Loads, Definition of Terms and Nomenclature

The following nomenclature and definition of load terms will apply to all internal structures unless otherwise noted:

3.8.3.3.1.1 Normal Loads

Normal loads are those loads to be encountered during normal plant operation and shutdown. They include the following:

- D - Dead loads or their related internal moments and forces, including any permanent equipment/system loads.

- L - Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure, and hydrostatic loads due to normal groundwater.

- T_o - Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

- R_o - Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

3.8.3.3.1.2 Severe Environmental Loads

Severe environmental loads are those loads that could infrequently be encountered during the plant life. Included in this category are:

- E - Loads generated by the Operating Basis Earthquake (OBE).

- W - Loads generated by the design wind and specified for the plant. (Wind load does not apply to internal structures.)

3.8.3.3.1.3 Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but are highly improbable. They include:

- E' - Loads generated by the Safe Shutdown Earthquake (SSE).

W_T - Loads generated by the Design Basis Tornado as specified in Section 3.3. They include loads due to the tornado wind pressure, loads due to the tornado-created differential pressures, and loads due to the tornado-generated missiles. (Tornado loads do not apply to internal structures.)

H - Hydrostatic loads due to maximum flood (as defined in Section 3.4).

3.8.3.3.1.4 Abnormal Loads

Abnormal loads are those loads generated by a postulated accident within a building and/or compartment thereof. Included in this category are the following:

P_a - Pressure equivalent static load within or across a compartment and/or building, generated by the postulated accident, and including an appropriate dynamic load factor to account for the dynamic nature of the load,

T_a - Thermal loads under thermal conditions generated by the postulated accident and including T_o.

R_a - Pipe reactions under thermal conditions generated by the postulated accident and including R_o.

A - Force on structure due to third level design margin requirement. (SMBDB)

Y_j - Jet impingement equivalent static load on a structure generated by the postulated accident, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_r - Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated accident, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m - Missile impact equivalent static load on a structure generated by or during the postulated break, such as pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios and as long as excessive deflections will not result in loss of function of any safety-related system.

3.8.3.3.1.5 Other Definitions

S - For structural steel. S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

The 33% increase in allowable stresses for steel due to seismic or wind loadings will not be used.

U - For concrete structures, U is the section strength required to resist design loads and based on methods described in ACI-349.

3.8.3.3.2 Internal Structure as Containment

No portion of the internal structure provides a direct containment function. The embedded part of the steel containment is designed such that it can withstand the design pressure without the assistance of the concrete walls.

3.8.3.3.3 Creep, Shrinkage and Local Stresses

No prestressed concrete design is considered for the design of the facility. Therefore, creep and shrinkage loads will be only considered to the extent they are provided in the reference concrete codes or as may be warranted by prudent design approach. The loads transferred from the support structure that generally influence local areas will be checked to insure that the local stresses are within acceptable limits to preclude impairment of the structural function.

3.8.3.3.4 Loads Due to Structural Margin Beyond Design Base (SMBDB)

Refer to CRBRP-3, Volume 1, Section 5.2 (Reference 10a of PSAR Section 1.6).

3.8.3.3.5 Sodium Fire Load

See Table 3.8-2 for the accident pressures and temperature loads.

3.8.3.3.6 Hot Sodium Spill Effect

The portions of the reactor cavity and cells, where exposure to radioactive hot sodium is a design basis accident, are provided with carbon steel liners designed to survive a sodium spill (see Section 3A.8). The liners will not compromise gas tightness of the cell.

3.8.3.3.7 Accident Temperature Load

See Table 3.8-2 for design temperatures.

3.8.3.3.8 Negative Pressure on the Liners

Any negative pressure on the liner will be resisted by a grid of structural anchors embedded in the concrete.

B. Load Combinations for Factored Load Conditions

For Factored Loads including earthquake (OBE or SSE), tornado (if applicable) and pipe break effects, etc., the following load combinations will be satisfied.

4) $1.6\ S = D + L + T_o + R_o + E'$

5) $1.6\ S = D + L + T_o + R_o + W_T$

6) $1.6\ S = D + L + T_a + R_a + P_a$

7) $1.6\ S^* = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E$

8) $1.7\ S^* = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E'$

9) $1.6\ S = D + L + T_o + R_o + H$

10) $1.6\ S = D + L + T_o + A$

In combinations (4) to (8) inclusive, thermal loads may be neglected when it can be shown that they are secondary and self-limiting in nature.

In combinations (6), (7) and (8), the maximum values of P_a, T_a, R_a, $Y_r + Y_j + Y_m$, including an appropriate dynamic load factor, will be used unless a time-history analysis is performed to justify.

Combinations (5), (7) and (8) will be first satisfied without the tornado missile load in (5) and without Y_r, Y_j and Y_m in (7) and (8). When considering these loads, however, local section strengths may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent will be checked for.

In loading combinations, no load factors of less than unity will be used in design or analysis.

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*For these two combinations, (7) and (8), in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 General Analysis Procedures

Structural analysis for each cell (except the reactor cavity) will be performed by considering one-foot wide vertical and horizontal strips through the structure. These strips, in essence, constitute structural segments which allow analysis by conventional methods.

The analysis of each cell will be performed by choosing one or more sections of one-foot width in both vertical and horizontal directions. The number of strips taken depends upon the configuration of each cell. In the case of analyzing a PHTS Cell, one typical vertical cross section (Figure 3.8-5) and two horizontal sections (Figures 3.8-6 and 3.8-7) are chosen for an illustration.

A rigid frame method of analysis will be used for determining the moments and shears in all members of a frame for each of the following anticipated loadings:

a) Dead and Live Loads,

b) Seismic Conditions,

c) Pressure Loads, and

d) Thermal Loads.

Loading combinations, using the factored loads as indicated in Section 3.8.3.3.10.1 will be used to establish the most critical design stresses for each component of the frame.

Vertical and horizontal strips thus analyzed will provide the necessary steel reinforcement for the vertical and horizontal directions of the particular cell analyzed.

Since the walls, ceilings and floors of each cell are considered as two-way slabs, the applied loads, used in the analysis, will be proportioned to the one-foot wide strips, in orthogonal directions, according to the ratio of their relative stiffnesses.

The cell design will be verified by using a three dimensional finite-element analysis with the computer program NASTRAN. The cell and adjacent structures will be represented in the mathematical model which will include the interaction with the containment shell and the exterior concrete wall. The
appropriate loads and load combinations will be used in the analysis. 33

Further detailed analysis will be performed in areas of load concentration and penetrations as noted in 3.8.3.4.3. The reactor cavity is treated as a hollow cylinder for structural analysis. When in-house computer programs are used, their correctness will be verified against acceptable published programs. All vertical loads will be transferred to the foundation mat by three principal structural elements, viz (a) walls of PHTS cells (b) perimeter wall around containment and (c) reactor cavity.

3.8.3.4.2 <u>Analysis for Seismic Loads</u>

Equivalent static seismic loads as developed from the dynamic analysis of the structure will be transferred through the horizontal slab diaphragms and vertical shear walls to the foundation mat. The details of seismic analysis are described in Section 3.7.

3.8.3.4.3 <u>Analysis for Openings</u>

Structural analysis will be performed around openings in walls and slabs
particularly where concentrated loads from thermal effects are induced. The
31 design will account for all the stresses in those areas and proper 28
reinforcement will be provided for the relief of such stress concentration.

3.8.3.4.4 <u>Liner Analysis</u>

The liner-anchor system will be designed and analyzed in accordance with the
requirements and criteria specified in paragraph 3.0 of PSAR Appendix 3.8-B.
37 Liner analysis is discussed in PSAR Section 3A.8.3.5.

3.8.3.4.5 Radiation Generated Heat Effect

Adequate heat removal capacity will be provided for the reactor cavity so that the radiation generated heat does not cause temperatures of the structural materials in excess of the ASME Code, Section III, Division 2 requirements. Where radioactive piping penetrates the concrete walls, adequate insulation or heat removal measures will be provided to control the temperature of structural materials within Code limits.

Radiation generated heat will be produced as a function of the position of the Reactor core with respect to the structures and the temperature distributions will be calculated. In PHTS cells, significant heat will be generated from other sources such as piping. In all such instances, adequate cooling capacity will be provided to limit the temperature of the structural materials.

3.8.3.4.6 Reinforcement Design

The reinforcing steel will be proportioned to meet the requirements of ACI-349. The bond and anchorage requirement of ACI-349 will be observed, since the interior structures primarily provide a confinement function. The reinforcement for each wall or slab will principally consist of a set of orthogonal bars on each face with additional reinforcement provided in areas of penetrations or load concentration.

3.8.3.4.7 Structural Steel Design

The structural steel components other than the cell liner systems and reactor vessel support ledge will be designed to the requirements of AISC specifications as identified in Section 3.8.2.2.1. When steel parts are stressed into the plastic range, an energy absorption check will be performed to assure the functional integrity.

3.8.3.4.8 Reactor Vessel Support Ledge Design

The reactor vessel support ledge will be designed in accordance with Subsection NF of the ASME Code, Section III, Division 1. The linear analysis method will be used. A finite element computer analysis will be performed for the support ledge.

The Steam Generator and Auxiliary Bays are designed to provide separation for
the three independent steam generator loops. The Steam Generator Bay consists
61 of three cells each having three main floor levels. Each cell contains one of
the three independent steam generating loops. The south and west side of the
structure are structurally connected to the Intermediate Bay and Diesel
Generator Building respectively. The structural system of this Bay is
essentially the same as the Intermediate Bay. The SGB gantry crane runway
61 rails at roof level are supported on the north and south walls of this Bay. 44
The gantry crane is capable of handling major equipment such as intermediate
pumps, evaporators and superheaters, and transferring them to the Maintenance
45 Bay which is located east of the Steam Generator Bay.

The portion of the Steam Generating System such as Water/Steam Circulating
System and the Steam Generator Auxiliary Heat Removal System (SGAHRS) is
located in the Auxiliary Bay. The exterior walls and the floor system of the
Auxiliary Bay are structurally the same as the Steam Generator Bay. The
interior walls of the Auxiliary Bay, as well as that of the Intermediate Bay
and the Steam Generator Bay, are of reinforced concrete with the exception of
the wall of the elevator shaft and Cell 202B walls which are of reinforced
block. The south side of the Auxiliary Bay is structurally connected to the
61 Steam Generator Bay.

The Maintenance Bay is the portion of the Steam Generator Building containing
a railroad siding and facilities for maintenance, cleaning, and laydown of
Steam Generator Building equipment. This bay is a Seismic Category I
structure but is not tornado-hardened. It is constructed of metal roof
47 decking and metal wall siding supported on structural steel beams and columns.

45

61 The overall approximate dimensions of the above four structures are as
follows:

		Inside Length (ft.)	Inside Width (st.)	Overall Height (ft.)
48 6	1. Intermediate Bay	260	Varies from 17' to 162'	124
	2. Steam Generator Bay	228	74	140
	3. Auxiliary Bay	228	30	153
48	4. Maintenance Bay	84	84	108

61 The top of the foundation mat for the Steam Generator Building, excluding the
Maintenance Bay, is at elevation 733' and grade is at elevation 815'. The
maintenance area as well as the laydown area and railroad track of the
Maintenance Bay is founded on competent rock. The laydown area and railroad
48 tracks are founded on Class "A" backfill.

See Section 1.2 for the Steam Generator Building General Arrangements and the general layout and configuration of the structures.

3.8.4.1.4 Diesel Generator Building

The Diesel Generator Building, located west of the emergency cooling towers, is a Seismic Category 1, tornado-hardened reinforced concrete structure that houses the Class 1E emergency diesel generators and diesel auxiliary equipment. The building is designed to allow separation of the three Class 1E and their associated power production and distribution equipment.

The exterior walls, interior walls and foundation mat of the building are of reinforced concrete. The Diesel Generator foundations are isolated from the rest of the structures to ensure that no vibrations from the diesel operations are transmitted to the rest of the structure. The roof slab is reinforced concrete supported by structural steel framing and the concrete walls. See Section 1.2 for the general layout and configuration of the structure.

3.8.4.1.4.1 Electrical Equipment Building

The Electrical Equipment Building (EEB) houses equipment and facilities used in the production of electrical power from the Emergency Diesel Generators. In addition, it houses the PHTS and IHTS sodium pump motor generators used to supply power to the IHTS and PHTS pumps in loop #3, and the switchgear and associated breakers for all IHTS and PHTS pumps. The Electrical Equipment Building is a part of the Nuclear Island complex. See Section 1.2 for the general layout and configuration of the structure.

An important function of the building is to provide an equipment removal path and routing area from the Nuclear Island Buildings to the Maintenance Shop and Warehouse via the TGB. To fulfill this function, a corridor approximately 26 feet wide, is located along the eastern part of the building. An equipment removal hatch 11'-0" x 15'-0" is provided to allow for removal of the largest piece of equipment from the SGB, CB or EEB through the hatch.

The floor elevation 794'-0" houses the emergency electrical power distribution equipment. Elevation 765'-0" houses the breakers for the PHTS and IHTS sodium pumps and 13.8 kv and 4.16 kv switchgear. The base elevation 733'-0" houses the PHTS and IHTS sodium pump motor generators for loop #3. Detailed equipment arrangements are shown on the Electrical Equipment Building Arrangement Drawings in Section 1.2.

3.8.4.1.5 Emergency Cooling Tower (ECT) Structure

The Emergency Cooling Tower Structure located approximately 700' north of the Reactor Containment Building, is a reinforced concrete structure which serves as a reservoir for the Emergency Plant Service Water (the basin) and will house the Emergency Plant Service Water pumps and the cooling tower units.

3.8.4.1.6 Diesel Fuel Storage Tank Foundation

Five diesel fuel storage tanks are located north and outside of the Diesel Generator Building. The diesel fuel tanks are buried and encased in a reinforced concrete mat which is founded on compacted Class A backfill material. For details, see Figure 3.8-1.

3.8.4.1.7 Electrical Manholes

Seismic Category 1 electrical manholes for duct bank carrying safety related cables are placed at various locations within the plant site. They are relatively small reinforced concrete structures. They will be founded on and surrounded by compacted Class A backfill, and will be located partially underground. An access opening in the top slab, at grade level, will be provided with a tornado missile shield cover.

3.8.4.1.8 Confinement Structure

The Confinement Structure is a reinforced concrete cylindrical enclosure with a spherical dome. The structure is located external to and concentric with, the containment vessel and is supported on the common Nuclear Island foundation mat. The overall dimensions are:

Cylindrical portion (from El. 733' to spring line) - I.D. = 196 feet, and thickness=4 feet.

Dome portion -- thickness=3 feet.

The structure functions as a tornado missile barrier, biological shield, and also as a protective enclosure against groundwater intrusion. It will be designed and constructed as a Seismic Category 1 structure.

43 | 32 | 3.8.4.1.9 <u>Interconnection of All Nuclear Island Seismic Category I Structures to the Reactor Containment Building</u> | 33

43 | The Seismic Category I building comprising the Nuclear Island will have a common foundation mat. The buildings surrounding the Reactor Containment will be connected to the confinement structure at all levels from the foundation to the roof, based upon the following considerations: | 33

1) The overall structural stability against lateral loads, particularly from hydrostatic and seismic forces, is greatly increased.

2) The distribution of lateral loads from the operating floor level down to the foundation mat is improved.

3) The **potential** of groundwater intrusion is eliminated.

4) The flexible joints or connections for piping and electrical systems at the interfaces between the Confinement Structure and other Category I buildings are eliminated. | 33

3.8.4.2 Applicable Codes, Standards and Specifications

The design and construction of all the Category 1 structures (except the containment vessel) are based upon the applicable sections of the following codes, standards, specifications and NRC Regulatory Guides.

A. American Concrete Institute (ACI)

ACI - 301-72	Specification for Structural Concrete for Buildings (Revised 1975)
ACI - 315-74	Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI - 349	Code Requirements for Nuclear Safety Related Concrete Structures
ACI - 347-68	Recommended Practice for Concrete Formwork
ACI - 305-72	Recommended Practice for Hot Weather Concreting
ACI - 211.1-74	Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete (Revised 1975)
ACI - 306-66	Recommended Practice for Cold Weather Concreting (Reaffirmed 1972)
ACI - 311-75	Recommended Practice for Concrete Inspection
ACI - 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI - 307-69	Specification for the Design and Construction of Reinforced Concrete Chimneys
ACI/ASCE-333	Tentative Recommendations for Design of Composite Beams and Girders for Buildings
ACI - 531-79	Building Code Requirements for Concrete Masonry Structures

B. American Institute of Steel Construction (AISC)

S310-1969	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings and Supplement No. S314.

C. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, (1974 Editions).

D. American Iron and Steel Institute

AISI Specification for the Design of Cold-formed Steel Structural Members, 1968 Edition with 1970, 1971 and 1972 supplements

E. American Society for Testing and Materials, ASTM Standards A-108 - Steel Bars, Carbon, Cold Finished Standard Quality E-84 - Surface and Burning Characteristics of Building Material

F. American Welding Society (AWSK)

AWS D.1.1-79 Structural Welding Code

AWS D12.1-1975 Reinforcing Steel Welding Code including Metal Inserts and Connections in Reinforced Concrete Construction

G. Crane Manufacturers Association of America, Inc., C.M.A.A. Specification No. 70.

H. American Railway Engineering Association (AREA)

I. Standard Building Code

J. American Association of State Highway and Transportation Officials (AASHTO) HB-11, Standard Specifications for Highway Bridges," 1973 Edition.

K. Code of Federal Regulations, Title 29, CFR1910, Occupational Safety and Health Standards, and Title 29 CFR1926, Safety and Health Regulations for Construction.

L. ERDA, Division of Reactor Research and Development, RDT Standards

F2-2 Quality Assurance Program Requirements
F2-4 Quality Verification Program Requirements

M. NRC Regulatory Guides

1.91 Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant sites.

1.115 Protection Against Low Trojectory Turbine Missiles. (1977)

See Sectio 3.8.3.2.3 for other applicable NRC Regulatory Guides.

Seismic analysis of the structures is covered in Section 3.7. The mathematical models will be as shown in the aforementioned section. Equivalent static seismic loads, as defined by the dynamic analysis, will be transferred through the horizontal slab diaphragms and vertical shear walls to the foundation mat.

Walls, floors and columns are the basic structural components of the buildings which will be analyzzed to carry and transfer the gravity and vertical loads to the common mat which is founded on rock. The lateral loads will be carried through horizontal diaphragm (floor) action to the vertical resisting elements (walls) depending upon their relative rigidity or stiffness. Torsional effect will be considered in the distribution of the lateral loads.

All Seismic Category 1 buildings of the nuclear island including the Confinement Structure, RCB, RSB (with the exception of the Radwaste Area), CB, DGB and SGB (with the exception of Maintenance Bay), will be on a common mat which is founded on rock.

The design and analysis of the common mat will be performed as described in Section 3.8.5.4.

3.8.4.4.2 Design Procedures

Design procedures will be in accordance with the applicable portions of the codes, standards and specifications listed in Section 3.8.4.2. The results derived in Section 3.8.4.4.1 will be used in design of structural steel and reinforced concrete.

Reinforced concrete structural elements will be designed by the strength method in accordance with ACI 349.

Structural steel frames or components of the buildings will be designed by the elastic analysis method in accordance with the provisions of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

Classical methods used in the design are standard textbooks, handbooks and publications as used in engineering practice.

The basic design approach for the structures is given in the following subsections.

3.8.4.4.3 Reactor Service Building

3.8.4.4.3.1 Reactor Service Area

This portion of the Reactor Service Building is designed as a Seismic Category 1 structure and will be analyzed as a multistory reinforced concrete structure made up of slabs, columns and walls. Interior walls that are required for shielding will be used as structural walls. Lateral loads will be resisted by shear walls with the floor and roof slabs acting as diaphragms.

The overhead crane runway rails will be supported on reinforced concrete brackets off the structural concrete walls. The crane will be designed as Seismic Category 1 equipment having such features as redundant receiving hoists and brakes. See Section 3.8.4.4.1 for the foundation design of the building.

3.8.4.4.3.2 Radwaste Area

The Radwaste Area of the RSB is an independent, seismic Category III designated structure. It consists of a steel framed structure above grade and of reinforced concrete slabs and walls at and below grade level and partially above grade (Solid Radwaste Area - on the east side). The Radwaste Area is supported by a reinforced concrete mat which is founded on sound siltstone with adequate bearing capacity.

The foundation for the west end of the Radwaste Area is not at grade elevation and is founded on compacted structural backfill.

The Radwaste Area structure is designed to meet the requirements of the Standard Building Code. In addition the structure below grade as well as the Solid Radwaste Area above grade are designed as a reinforced concrete structure.

The upper part of the Radwaste Area, the steel framed structure is designed to ensure that the adjacent seismic Category 1 structure of Reactor Service Area is not damaged nor its safety functions compromised during an SSE.

3.8.4.4.4 Control Building

As described in Section 3.8.4.1.2, the Control Building is structurally connected to the Diesel Generator Building and the Intermediate Bay of the Steam Generator Building. The building is a box-type structure with exterior walls of reinforced concrete and intermediate floors and roof of composite construction made up of reinforced concrete slabs on structural steel framing. Interior walls are of reinforced concrete and reinforced concrete masonry block. The interior structural steel columns are designed to carry vertical loads while the exterior walls, roof and intermediate floors are designed to resist vertical as well as lateral loads.

See Section 3.8.4.4.1 for the foundation design of the building.

3.8.4.4.5 <u>Steam Generator Building</u>

As described in previous Section 3.8.4.1.3, the Steam Generator Building is
structurally connected to the Confinement Structure, Diesel Generator Building
and the Control Building. The intermediate floors and the roof of the
Intermediate, Steam Generator and Auxiliary Bays will consist of composite
structural design (structural steel framing and reinforced concrete slabs) on
structural steel columns designed to carry the vertical loads. The
Maintenance Bay will be designed as a space frame with beams and columns
resisting moments and shear loads. The intermediate floors and the roof, in
61 conjunction with reinforced concrete exterior and interior walls, will resist
the vertical and lateral loads.

The gantry crane, which is Seismic Category I, is supported on the roof of the
45 Steam Generator and Maintenance Bays and the Diesel Generator Building.

As described in Section 3.8.4.4.1, the reinforced concrete mat for the
buildings will be founded on the rock.

3.8.4.6.2.3 Structural and Miscellaneous Steel

For all material used as structural steel, mill test reports giving chemical composition and physical properties will be obtained for approval. Fabrication and erection will be in accordance with AISC specifications and Section III of ASME Code.

Inspection will be conducted at the fabrication plant as well as in the field. Welding inspection will be as outlined in Burns and Roe's Structural, Materials, and Construction Specifications.

3.8.4.6.3 Special Construction Techniques

The structures will be constructed using normal construction methods and techniques.

3.8.4.7 Testing and In-Service Surveillance Requirements

There are no testing and in-service surveillance requirements for the structures.

3.8.4.8 Masonry Walls

See Appendix 3.8-D.

3.8.5 Foundation and Concrete Supports

3.8.5.1 Description of the Foundation and Supports

3.8.5.1.1 General Description

The foudation for the following Seismic Category 1 structures consists of a combined reinforced concrete mat laid out to envelope these structures. See Figure 3.8-2 for the mat layout.

(a) Reactor Containment Building (RCB)
(b) Confinement Structure
(c) Reactor Service Building (RSB), excluding Radwaste Area
(d) Control Building (CB)
(e) Diesel Generator Building (DGB)
(f) Steam Generator Building (SGB), excluding the Maintenance Bay

The thickness of the combined mat based upon the stress and stability considerations is 18' except under the RCB where it is 15'. The mat slab rests upon a firm rock strata and its bottom is located at El. 715 which is 100 feet below the finished grade. A fill slab of suitable design is placed over the RCB portion of the mat.

The Emergency Cooling Tower Structure will be supported by a reinforced concrete mat founded on competent rock. The SGB Maintenance Bay foundation will be on competent rock. The diesel fuel oil storage tank foundation, the Emergency Plant Service Water System supply and return headers in the yard and the underground Class 1E electrical ducting and Category 1 pipe will be founded on and surrounded by the compacted structural backfill.

3.8.5.1.2 Design Features

The combined mat concept for the Seismic Category I structures noted in subsection 3.8.5.1.1 rests upon two basic considerations, namely; (a) to reduce seismic responses at component supports, and (b) to minimize buoyancy effect due to maximum flood on relatively lighter portions of the foundations by combining them with heavier portions.

Several provisions will be included to prevent the intrusion of groundwater or 33
61 flood water into the steel containment. They are:

a. A continuous membrane waterproofing system along all external building faces to grade.

b. A thick reinforced concrete structure designed for crack control with continuous waterstops provided at all external construction joints below plant grade.

61

For Seismic Category I buildings, appropriate drainage systems, where required, will be provided to dispose of any potential intrusion of groundwater. Flood water protection provisions are discussed in Section 3.4.1 of this PSAR.

3.8.5.1.3 Load Transfer

The loads from the superstructure will be transferred to the foundation mat via structural elements such as columns and walls. The load transfer structures will have necessary configurations, and strength to meet the shielding as well as structural requirements.

Since the mat is founded upon competent rock, no relative local subsidence is expected. Therefore, no adverse effect due to this factor will be considered in the foundation design.

The mat analysis will provide for all interface loads arising due to the interaction between the mat and connected structural elements such as walls and columns.

3.8.5.1.4 Large Equipment Supports

The Reactor Vessel is supported on a steel ledge, partially embedded in the RV cavity walls. All vertical and lateral forces on the RV will be transferred
61 through the ledge to the cavity wall and the foundation mat.

TABLE 3.8-1b

Loading Combinations for Airlocks

The loading combinatios for which the airlocks shall be designed are as follows:

Testing:	D + Tt + Pt
Normal:	D + L + To + OBE
Accident and Environmental:	D + L + T' + Pl + OBE
	D + L + T' + Pl + SSE
	D + L + T' + Pe + OBE
	D + L + T' + Pe + SSE
	D + L + SSE

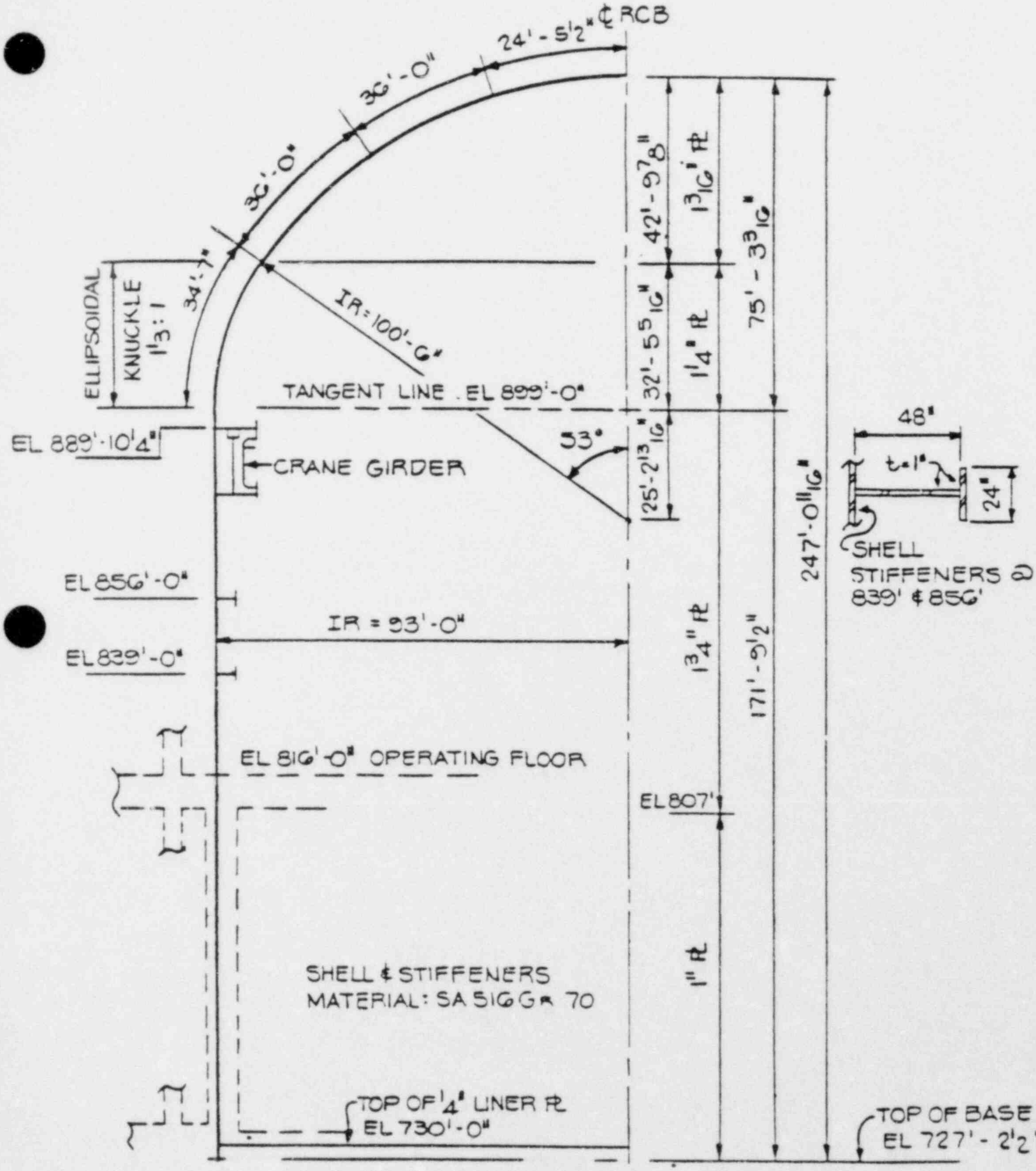

FIGURE 3.8-3 CRBRP CONTAINMENT VESSEL

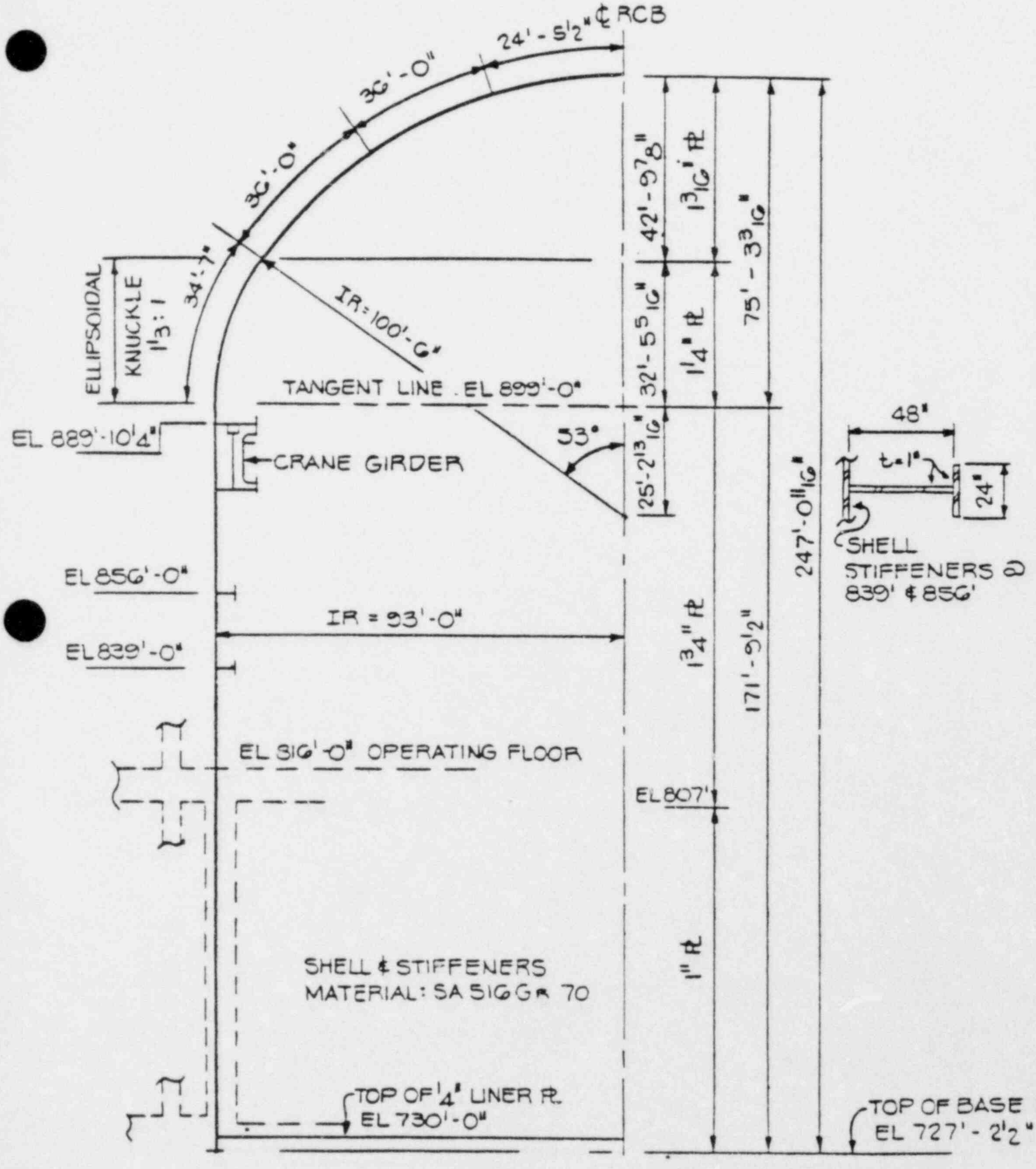

FIGURE 3.8-3 CRBRP CONTAINMENT VESSEL

FIGURE 3.8-4

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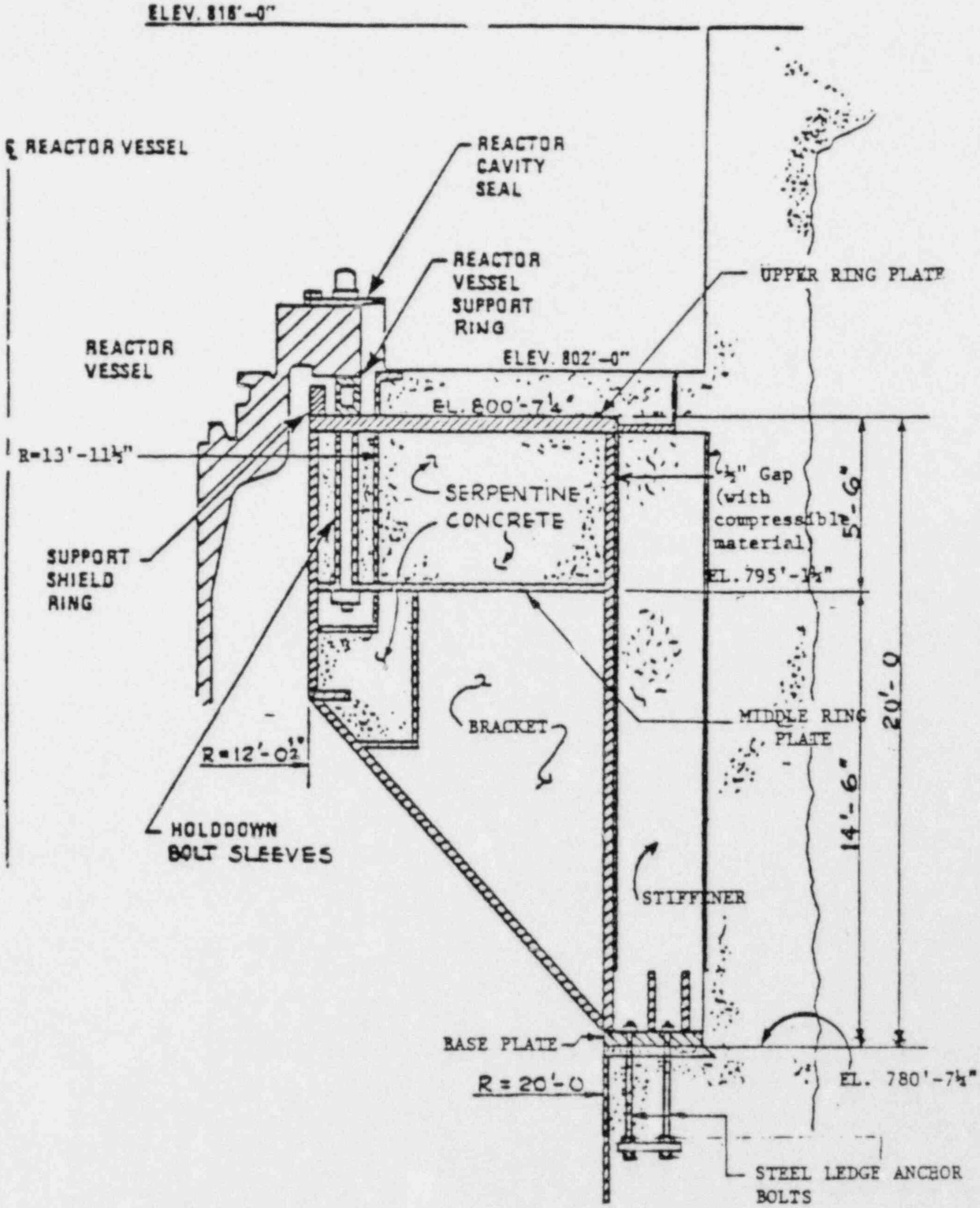


FIGURE 3.8-9 TYPICAL SECTION THROUGH REACTOR VESSEL LEDGE SUPPORT (Sheet 1 of 2)

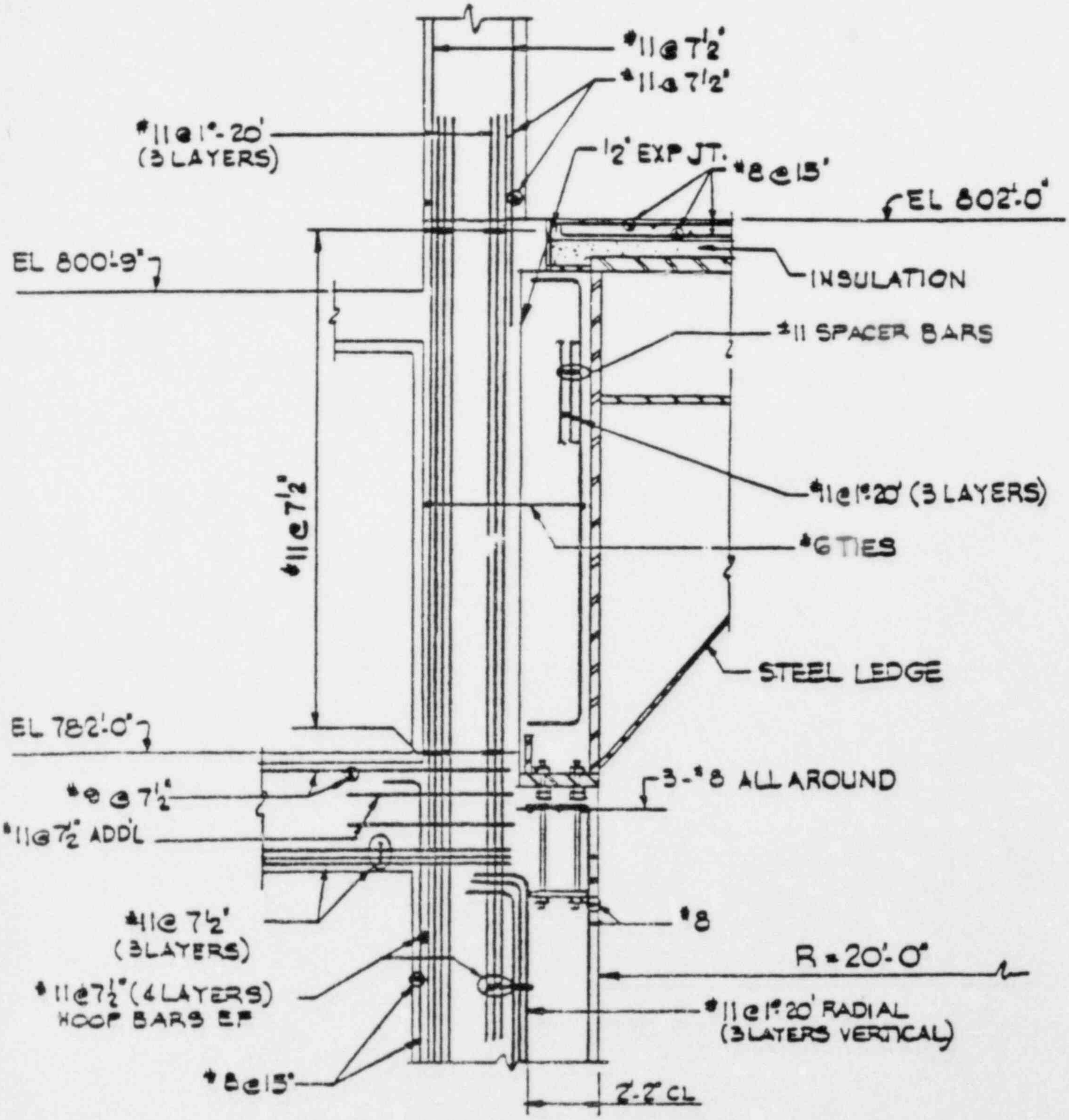
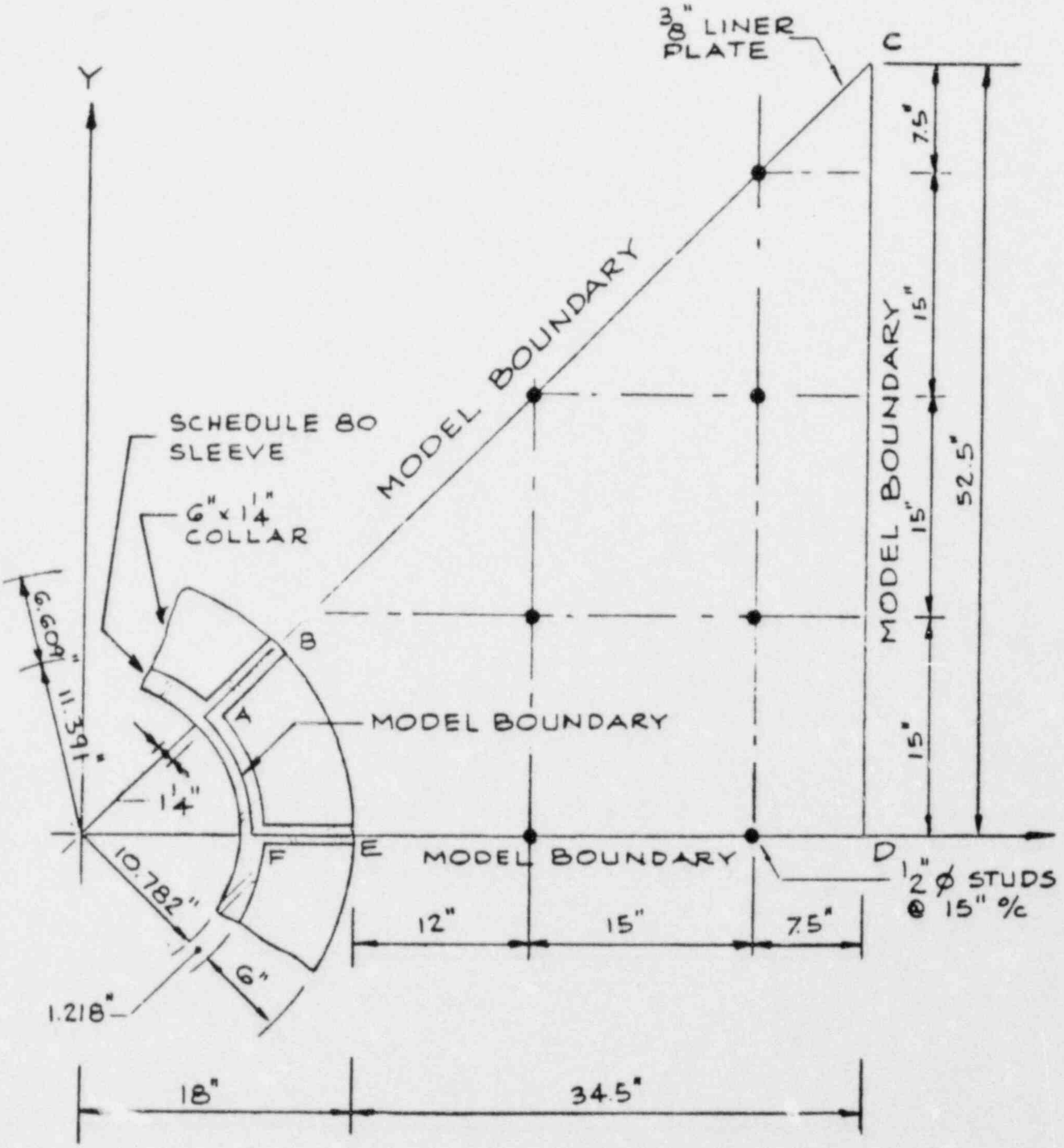


FIGURE 3.8-9 TYPICAL SECTION THROUGH REACTOR VESSEL LEDGE SUPPORT
(Sheet 2 of 2)



Sectional Elevation Of Wall Liner Penetration

Figure 3A.8-8

Amend. 70
Aug. 1982

The deformation controlled stresses and strains were determined by a 2-D axisymmetric model of a cross section of the ring. The model used for both the thermal and thermal stress analyses of the ring is shown in Figure 4.2-77.

Thermal boundary conditions applied to the upper core former model were fluid temperatures applied through a convection coefficient in several different regions of the model as shown in the figure. The structure was analyzed elastically for the U-1b, U-2b, U-18 and E-16 thermal events which may be conservatively used to umbrella all other loadings.

All regions of the CFS were shown to be adequate using elastic analysis methods except the top surface of the upper ring. This area was shwon to be adequate by simplified inelastic methods. The fatigue damage at this location was .414 with a creep damage of .239. This combination of
59 damages falls within the creep fatigue interaction envelope of Code Case 1592.

59 4.2.2.4.2 Upper Internals Structure

This section presents the analysis performed in support of the final design of the Upper Internals Structure (UIS) and used to demonstrate the adequacy of this component for the expected service conditions and environment. The adequacy of the design is based primarily upon meeting the criteria of Section III of the ASME Boiler and Pressure Vessel Code including Code Case N-47, and supplemented by RDT standards F9-4T and F9-5T and special project structural design rules. A summary of the components analyzed, material properties, structural design criteria, mechanical loads, thermal environment, methods of analysis, and structural analysis is presented herein.

59 4.2.2.4.2.1 Components Analyzed

The major components of the UIS are identified in Figure 4.2-45. A brief outline of the functions of the UIS is given in Section 4.2.2.2.1.7. A list of the components of the UIS analyzed to demonstrate structural adequacy of the design are:

- o Lower Plate and Ligament
- o Upper Plate
- o Support Columns
- o Shear Webs
- 51 o Core Barrel Key

- o Instrumentation Posts
- o Upper and Lower Shroud Tubes
- o Chimney Assemblies
- o Mixing Chamber Thermal Liners
- o IVTM Port Plug
- o IVTM Port Plug Cover

4.2.2.4.2.2 Material Properties

The ASME Code is the prime source for materials properties. For material properties not specified in Section III of the code or applicable code cases, the mechanical properties are based on the Nuclear Systems Materials Handbook TID-26666, RDT Standard F9-4T, RDT Standard F9-5T or the sources given in Section 4.2.2.3.3.

There are no irradiation effects on the mechanical properties of Inconel 718 or 316 stainless steel at the highest fluence levels attained at the end of its 30 year service life. The maximum fluence level is less than:

$$1 \times 10^{21}\ n/cm^2$$

Type 316 stainless steel is a non-age hardenable alloy. Therefore, no significant changes in strength or hardness should result from long term exposures at temperatures up to 1100°F. Inconel 718 is an age hardenable alloy, however, the age hardening process does not result in significant reductions in mechanical properties when subjected to temperatures up to 1100°F for component lifetimes. No allowances have been made for the effects of thermal aging on the properties of either alloy. However, experimental material properties programs to study the behavior of both alloys due to the thermal environment and sodium exposure are discussed in Section 1.5.

The sodium effects of Section 4.2.2.3.3.2.1 are implemented in creep-fatigue damage evaluations of 316 stainless steel.

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Cyclic Hardening

A factor K_c is applied to alter the stress-strain curve yield surface.

where:

$$K_c = (-0.144 + 3.094\,\Delta\epsilon)^{1/2} \text{ for } \Delta\epsilon > 0.36\%$$

If the strain range $\Delta\epsilon$ is $\leq 0.36\%$, then $K_c = 1.0$.

High Cycle Design Fatigue Strength

Stainless steel materials subjected to high cycle thermal fluctuations and flow induced vibration phenomena require a fatigue strength evaluation beyond the Code Case 1592 (N-47) curve limit of 10^6 cycles. The Code Case 1592 (N-47) curve is extrapolated beyond 10^6 cycles using a slope on cycles of -0.12 for load controlled situations. In cases where conditions are strain controlled, the special purpose high-cycle fatigue criterion, as described in 4.2.2.3.2.3, is used beyond 10^6 cycles.

4.2.2.4.2.3 Structural Design Criteria

The portion of the UIS within the reactor vessel operates at elevated temperatures above 800°F. Under these circumstances the UIS is classified as an elevated temperature structure and is designed and analyzed as an ASME - III Code Class 1 component.

Alternate structural design criteria have been adopted in the cumulative creep-fatigue damage rules of Code Case 1592 (N-47) and RDT Standard F9-4. These criteria assume that in compressive hold, creep rupture damage is 20% as damaging as the damage caused by the same sustained stress in tension. It applies for austenitic stainless steel (Types 304 and 316) at metal temperatures less than 1200°F (649°C). At times in the duty cycle when sustained stresses are tensile, damage is computed in accordance with Code Case 1592 (N-47).

5.5.3.3 Pump Characteristics

The recirculation pump in each loop draws water from the steam drum and forces the water through the two evaporators and back to the steam drum. The pump is a single-stage centrifugal pump, which delivers 2.22 X 10^6 pounds per hour of water to the two evaporators. The pump provides a pressure head of about 148 psi at the plant design conditions. This flow provides a quality of 50% at the exit from the steam generator evaporator units at the plant design power. Since the pumps are constant speed, the evaporator exit steam quality decreases as plant power is reduced.

During normal plant operation, feedwater is mixed with saturated water in the steam drum providing subcooled water to the recirculation system. This subcooled water results in an adequate net positive suction head (NPSH) for the pump during normal operation. Under certain operating conditions, however, the recirculation pumps will be operating when there is essentially no feedwater being delivered to the steam drum, such as during start-up before there is any significant power from the reactor, and, in this case, an adequate NPSH for the recirculation pump is provided by the elevation difference between the steam drum and the pump.

The shaft seal of the recirculation pump is externally supplied with filtered injection water from the main feedwater system which cools the seal and minimizes accumulation of corrosion particulates which could otherwise cause accelerated seal wear and increased leakage. In addition, the recirculation pump has an internal auxiliary impeller and cooler to provide seal cooling in the event that the external seal injection system is lost. However, the internal seal cooling system only functions when the recirculation pump is running. With external seal injection operating, 3.5 gpm of injection water is added to each recirculation loop. With external seal injection secured, there is a net loss of 1.5 gpm of recirculation water lost from each loop through the recirculation pump seal staging valve.

In the unlikely event that the recirculation pump trips and external seal injection is lost simultaneously, the pump shaft seal will overheat and could fail sufficiently to result in a large increase in seal leakage. If not controlled and if the SGAHRS system is initiated, this leakge could potentially compromise the Protected Water Storage Tank (PWST) 30 days capacity. Accordingly, a valved bypass line is provided around the recirculation pump and its inlet and outlet isolation valves. The bypass line is designed to provide hydraulic resistance equivalent to that of the stalled recirculation pump and its connected piping and valves inside the bypass connections. Therefore, the flow of water under natural circulation conditions to the evaporators through the bypass line will be essentially the same as through the normal path.

In the event of a loss of offsite power, both the recirculation pump and main feed pumps will trip. In this case, the recirculation pump will be isolated and bypassed automatically because the arrangement of the valve actuator power supplies causes the isolation and bypass valves to realign to their fail safe positions. For other SGAHRS events, the recirculation pump trips and seal injection flow stops, the operator will attempt to restore seal cooling (e.g., by starting the startup feed pump) before the recirculation pump seal overheats. If these attempts are not successful, the operator will activate the recirculation pump bypass line from the main control panel by opening the bypass valves and shutting the recirculation pump isolation valves within one hour of the loss of seal cooling. The 30 day PWST water supply is therefore ensured since PWST inventory is sufficient to allow up to one and a half hours for operator action. For SGAHRS events where all the recirculation pumps continue to operate but seal injection flow can not be provided, the operator has over ten hours to activate the recirculation pump bypass lines to ensure PWST inventory is not jeopardized as a result of recirculation pump seal staging flow.

5.5.3.4 Valve Characteristics

A number of isolation valves are used in the steam generation system to permit equipment to be isolated from the system for maintenance. These valves are conventional valves which are normally open during operation and are closed only to permit maintenance or repair of the equipment.

Quick-operating isolation valves are located on the inlet lines of the evaporator units and the superheater unit to permit isolation of a unit from the rest of the system in case of a failure in the water or steam to sodium barrier, resulting in a sodium-water reaction. These isolation valves will close in about four seconds after an automatic or manual signal. Isolation of a failed unit will limit the quantity of water or steam available for the sodium-water reaction and, in the case of evaporator units, permit dumping of water and steam to a dump system to further reduce the magnitude of the sodium-water reaction.

The evaporator water dump valve is a quick opening shut-off valve located on the inlet line of each evaporator unit between the evaporator and the evaporator inlet isolation valve. This valve will open at the same time the evaporator isolation valves are being closed. This valve dumps water and steam from the evaporator unit to reduce the total quantity of water available for a sodium-water reaction and to reduce the pressure in the unit. This valve is held open until the water-steam pressure in the unit drops to approximately 250 psi, when the valve is automatically closed. Inert gas is then supplied to the unit to reduce the possibility of sodium and sodium-water reaction products entering the water-steam side of the steam generation system.

TABLE 5.5-2

MANDATORY CODE CASES FOR SGS AS APPLICABLE

Code Case	
1473-1	Short Time High Temperature Service for Section VIII, Division 2 -Modifications to Section VIII, Division 2, are provided for vessels which are to operate during part of their service life (less than 2500 Hrs.) at temperatures above those now provided for in Section VIII, Division 2.
1481	Elevated Temperature Design of Class 2 and 3 Nuclear Components -Modifications to Section III are provided for Class 2 and 3 components with normal operating temperatures above those provided for Section III.
1489	Elevated Temperature Design Section VIII, Division 2 -Allowances for elevated temperature design are provided for in Section VIII, Division 2.
1592	Components in Elevated Temperature Service Section III, Class 1.
1593	Fabrication and Installation of Elevated Temperature Components, Section III, Class 1.
1594	Examination of Elevated Temperature Nuclear Components, Section III, Class 1.
1595	Testing of Elevated Temperature Nuclear Components, Section III, Class 1.
1596	Protection Against Overpressure of Elevated Temperature Components Section III, Class 1.
1606	Stress Criteria Section III Classes 2 and 3, Piping Subject to Upset, Emergency, and Faulted Operating Conditions. -Design criteria are provided for Class 2 and 3 piping subject to upset, emergency, and faulted conditions.
1607	Stress Criteria Section III, Class 2 and 3, Vessels Subject to Upset, Emergency, and Faulted Operating Conditions. -Design criteria are provided for Class 2 and 3 vessels subject to upset, emergency, and faulted conditions.

TABLE 5.5-3

SGS EQUIPMENT LIST AND MATERIAL SPECIFICATIONS

Component	Material
Major Components:	
Superheaters	2-1/4 Cr-1Mo
Evaporators	2-1/4 Cr-1Mo
Steam Drums	SA-516CS/SA299
Recirc. Pumps	Carbon Steel
Major Subsystem Components:	
Sodium Rupture Discs	Inconel 600
Reaction Prod. Sep. Tank	Carbon Steel
Centrifugal Separator and	Carbon Steel
Drain Tank	Carbon Steel
Flare Stack w/Igniter	Carbon Steel
SWRPRS Vent Line Rup. Disc	Carbon Steel
Evap. Water Dump Tank	Carbon Steel
Sodium Dump Tank	Carbon Steel
Sodium Dump Tank Rupture Disc	*
Piping and Headers:	
S.G. Subsystem and Feed-water subsystem:	
SGB Wall to Main Fdwtr. Drum Isolation Valve	SA-106, Gr B
Main Fdwtr. Drum Isolation Valve to Steam Drum	SA-106, Gr B
Startup Feedwater Control Valve Piping	SA-106, Gr B
Drum to Pump Suction Header	SA-106, Gr B
Pump Suction Header	SA-106, Gr B
Pump Suction Header to Pump	SA-106, Gr B
Pump to Pump Discharge Tee	SA-106, Gr B
Pump Discharge Tee	SA-106, Gr B
Pump Discharge Tee to Evap. Inlet Isolation Valve	SA-106, Gr B
Evap. Inlet Isolation Valve to Evap.	SA-106, Gr B
Evap. to Drum	SA-106, Gr B
Drum to Superheater	SA-106, Gr B
S. H. to S. H. Isolation Valve	1-1/4 Cr-1/2Mo
S. H. Isolation Valve to SGB Wall	2-1/4 Cr-1 Mo
S. H. Bypass Inlet Piping	SA-106, Gr B
S. H. Bypass Outlet Piping	1-1/4 Cr-1/2 Mo
Recirculation Pump Bypass	SA-106, Gr B

TABLE 5.5-3 (Continued)

	Component	Material
	SWRPRS:	
	Sodium Rupture Discs Discharge Lines to Reaction Products Separation Tank	Carbon Steel
	Sep. Tank to Centrif. Sep.	Carbon Steel
	Centrif. Sep. to Flare Stack	Carbon Steel
	Reaction Prod. Sep. Tank Equilizer Line	Carbon Steel
41	Cent. Sep. Drain Line	Carbon Steel
42	Sodium Dump Subsystem (Piping):	
	Sod. Dump Tank Vent Line to Stack	SA-106, Gr B
	Equilizer Gas Line to Expansion Tank	304 SS
41		
	Water Dump Subsystem and Relief Lines (Piping):	
	Stm. Drum Relief Valve Inlet	SA-106, Gr B
41	Stm. Drum Relief Valve Discharge	SA-106, Gr B
	S.H. Relief Valve Inlet	1-1/4 Cr-1/2Mo
46	S.H. Relief Valve Discharge	1-1/4 Cr-1/2Mo
	Evap. Relief Valve Inlet	SA-106, Gr B
	Evap. Water Dump Valve Inlet	SA-106, Gr B
	Evap. Water Dump Valve Discharge	SA-106, Gr B
	Evap. Relief Valve Discharge	SA-106, Gr B
41	Water Dump Tank Discharge	SA-106, Gr B
	Na-H_2O Leak Detection Subsystem (Piping):	
47 42	Na Supply and Return Piping	304 SS
	Drum Blowdown (Piping):	
41	Drum to SGB Wall	SA-106, Gr B
	SGAHRS (Piping):	
	Stm. Supply Valves to SGAHRS F.W. Pump	SA-106, Gr B
	Stm, Supply Valve to SGAHRS HX	SA-106, Gr B
41	Water Return Valve from SGAHRS HX	SA-106, Gr B
	AFW Supply Valve to Main Fedwtr. Line	SA-106, Gr B

42

TABLE 5.5-3 (Continued)

Valves:

S.G. Subsystem and Feedwater:

S.H. Inlet Isolation Valve	Carbon Steel
S.H. Outlet Isolation Valve	2-1/4Cr-1Mo
S.H. Outlet Check Valve	2-1/4Cr-1Mo
Recirc. Pump Bypass Isolation Valve	Carbon Steel
Evap. Inlet Isolation Valve	Carbon Steel
Evap. Outlet Check Valve	Carbon Steel
Pump Suction Isolation Valve	Carbon Steel
Main F.W. Control Valve	Carbon Steel
Startup F.W. Control Valve	Carbon Steel
Main F.W. SGB Isolation Valve	Carbon Steel
Main F.W. Check Valve	Carbon Steel
Main F.W. Isol. Valve Drum Isol. Valve	Carbon Steel
Superheater Bypass Valve	Carbon Steel

Safety Relief Valves:

Steam Drum (Safety Function Only)	Carbon Steel
Evaporator	Carbon Steel
Superheater	2-1/4Cr-1Mo

SWRPRS:

SWRPRS Vent Line Check Valve	Carbon Steel
SWRPRS Atmospheric Seal Bypass Valve	*

Water Dump Subsystem:

Evap. Water Dump Valve	Carbon Steel
Water Dump Tank Drain Valve	Carbon Steel

Sodium Dump Subsystem:

Sodium Dump Tank Cover-Gas Relief Valve	*

Drum Drain:

Drum Drain Isolation Valve	SA-106, Gr B

TABLE 5.5-5
SGS PUMP AND VALVE DESCRI[illegible]ION

PUMPS	ACTIVE	INACTIVE	ACTUATING SIGNAL
Recirculation Pump		X	N/A
VALVES			
Recirc. Pump Suction Isolation	X		Loss of Offiste Power or Manual** (Remote)
Recirc. Pump Bypass Isolation	X		SWRPRS or Manual (Remote)
Evaporator Inlet Isolation	X		Loss of Offsite Power, SWRPRS**, or Manual (Remote)
Evaporator Inlet Water Dump		X	SWRPRS
Evaporator Outlet Relief	X		SWRPRS**, High Pressure-SWRPRS** or High Pressure Evaporator (Steam)
Steam Drum Relief	X		High Pressure - Steam Drum
Superheater Inlet Isolation		X	SWRPRS
Superheater Relief	X		SWRPRS**, High Pressure SWRPRS** or High Pressure Superheater (Steam)
Superheater Outlet Isolation	X		SWRPRS**, OSIS or Low Superheater Outlet Pressure
Superheater Bypass Valve	X		SWRPRS**, OSIS, or Low Superheater Outlet Pressure
Steam to SGAHRS HX		X	Manual (L.O.)*
Water from SGAHRS HX		X	Manual (L.O.)*
Steam to SGAHRS Auxiliary FW Pump		X	Manual
Feedwater from SGAHRS		X	Manual (L.O.)*
Main Feedwater SGB Isolation	X		SWRPRS**, High Steam Drum Level, Low Steam Drum Pressure, Cell Temp and Humidity
Main Feedwater Drum Isolation		X	High Steam Drum Level
Main Feedwater Check Valve		X	Simple Check
Main Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
Startup Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
Evaporator Outlet Check Valve		X	Check Valve
Superheater Outlet Check Valve		X	Check Valve
Steam Drum Drain Isolation	X		SWRPRS**, SGAHRS Initiation, Low Steam Drum Pressure

* L.O. - Locked open
** This function is not safety active

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TABLE 5.5-5 (Continued)

	Valves	ACTIVE	INACTIVE	ACTUATING SIGNAL
	SWRPRS Stack Check Valve		X	Check Valve
	SWRPRS Atmospheric Seal Bypass		X	Manual
41	Sodium Dump Tank Pressure Relief		X	High Sodium Dump Tank Pressure
59	Evaporator Water Dump Tank Drain		X	Manual

42

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Dec. 1980

TABLE 5.5-6

SGS LOADING CONDITIONS

35 ASME III Code Class 3 system components will be designed considering the following load combinations:

1 <u>Pumps (Recirculation Loop)</u>

Operating Condition	Component	Load	Stress Limit
See Note 1	Pump Case	Design Pressure	Section III
		Design Temperature	Allowable Stress
	Cover	Design Pressure	Section III
	Bolting	Design Temperature	Allowable Stress
		Safe Shutdown Earthquake	
		Pump Thrust	
		Weight	
		Gasket Loads	

Note 1: Design pressures and temperatures of the recirculation system components are established using pressures and temperatures occurring during emergency and faulted transients. The design temperature is not exceeded during these transients. The design pressure may be exceeded by not more than 10% during these transients. Normal and upset conditions are not controlling.

2 <u>Valves (Recirculation Loop and Main Water/Steam)</u>

35 The valve pressure retaining parts designed to ASME - III Class 3 will withstand seismic forces and pipe loads of the SSE as well as design pressure and temperatures. On other parts, if earthquake needs are to be considered, the following applies:

Operating Condition	Loads
Upset	1. Normal Operating
	2. OBE
Faulted	1. Normal Operating
	2. SSE

TABLE 5.5-7

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
1. Steam Generator Subsystem & Feedwater Subsystem					
SGB Wall to Drum Feedwater Isolation Valve	10", sch. 160	1	3	3	3000 psig, 500 F
Feedwater Drum Isolation Valve to Steam Drum	10", sch. 140	1	3	3	2200 psig, 650 F
Drum to Pump Inlet Header	10", sch. 140	4	12	3	2200 psig, 650 F
Pump Headers (Inlet)	18", sch. 140	1	3	3	2200 psig, 650 F
Pump Inlet Header to Pump	18", sch. 140	1	3	3	2200 psig, 650 F
Pump to Pump Discharge Tee	12", sch. 160	1	3	3	2450 psig, 650 F
Pump Discharge Tee	12", sch. 160	1	3	3	2450 psig, 650 F
Pump Discharge Tee to Evaporator Isolation Valve	10", sch. 160	2	6	3	2450 psig, 650 F
Evaporator Isolation Valve to Evaporator	10", sch. 160	2	6	3	2400 psig, 650 F
Recirculation Pump Bypass	8", sch. 160	1	3	3	2400 psig, 650 F
Evaporator to Drum	16", sch. 140	2	6	3	2200 psig, 650 F
Drum to S.H.	12", sch. 140	1	3	3	2200 psig, 650 F
S.H. to Isolation Valve	16", sch. 160	1	3	3	1900 psig, 935 F
S.H. Bypass Inlet Line	4", sch. 160	1	3	3	2200 psig, 650 F
S.H. Bypass Outlet Line	4", sch. 160	1	3	3	1900 psig, 935 F
Isolation Valve to SGB Wall	16", sch. 160	1	3	3	1900 psig, 935 F
Startup Feedwater Control Valve Piping	4", sch. 160	1	3	3	3000 psig, 500 F
2. SWRPRS					
Sodium Rupture Disc Discharge Lines to Separator Tanks	18" nom., var. wall, 24" nom., var. wall, 26.46" nom., 2.2" wall	3	9	3	300 psig, 800 F
Separator Tanks to SGB Roof	16", sch. 40	1	3	3	125 psig, 200/800 F
SGB Roof to Flare Tip	16", sch. 40	1	3	ANSI B31.1	125 psig, 100°F
Recirculation Pump Bypass	8", sch. 160	1	3	3	2400 psig, 650F

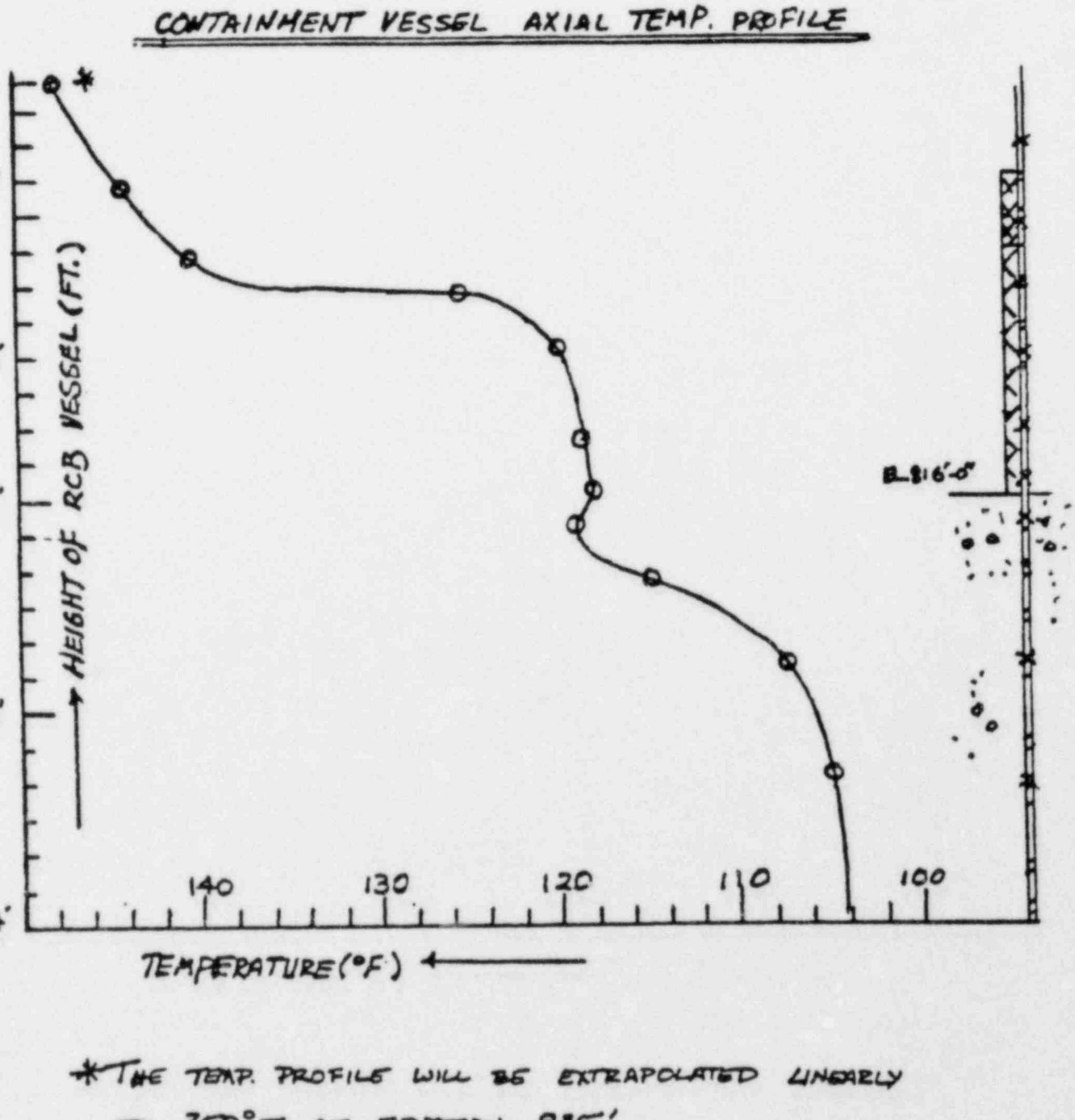

FIGURE 6.2-11

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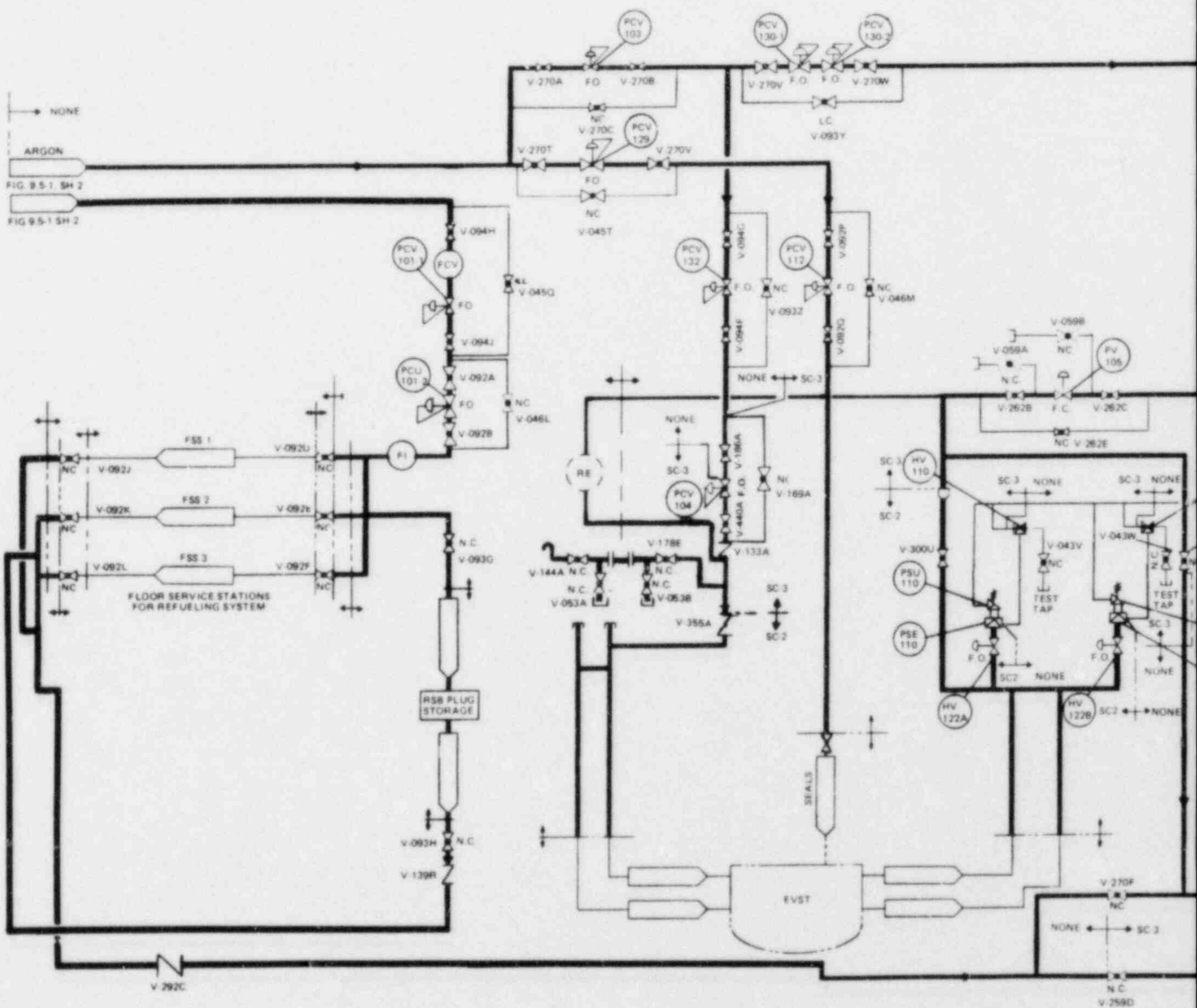

NONE
ARGON
FIG. 9.5-1, SH 2
FIG 9.5-1 SH 2
PCV 103
V-270A
F.O.
V-270B
NC
V-270C
PCV 129
V-270T
V-270V
NC
V-045T
PCV 130-1
PCV 130-2
V-270V
F.O.
F.O.
V-270W
LC
V-093Y
V-094H
PCV 101-1
PCV
FO
V-045Q
V-094J
PCU 101-2
V-092A
FO
NC
V-046L
V-092B
FI
PCV 132
V-094C
F.O.
NC
V-093Z
V-094F
PCV 112
V-092P
F.O.
NC
V-046M
V-092G
NONE
SC-3
RE
NONE
SC-3
PCV 104
V-186A
V-440A
NI
V-169A
V-133A
FSS 1
FSS 2
FSS 3
V-092J
V-092U
V-092K
V-092E
V-092L
V-092F
NC
FLOOR SERVICE STATIONS FOR REFUELING SYSTEM
N.C.
V-093G
RSB PLUG STORAGE
V-093H
N.C.
V-139R
V-178E
V-144A
N.C.
V-053A
V-053B
V-355A
SC-3
SC-2
SEALS
EVST
V-292C
V-059B
NC
V-059A
N.C.
PV 105
V-262B
F.C.
V-262C
NC
V-262E
SC-3
HV 110
SC-2
SC-3
NONE
V-043V
NC
V-043W
V-300U
PSU 110
TEST TAP
PSE 110
F.O.
SC2
NONE
HV 122A
HV 122B
SC2
NONE
NONE
V-270F
NC
NONE
SC-3
N.C.
V-259D

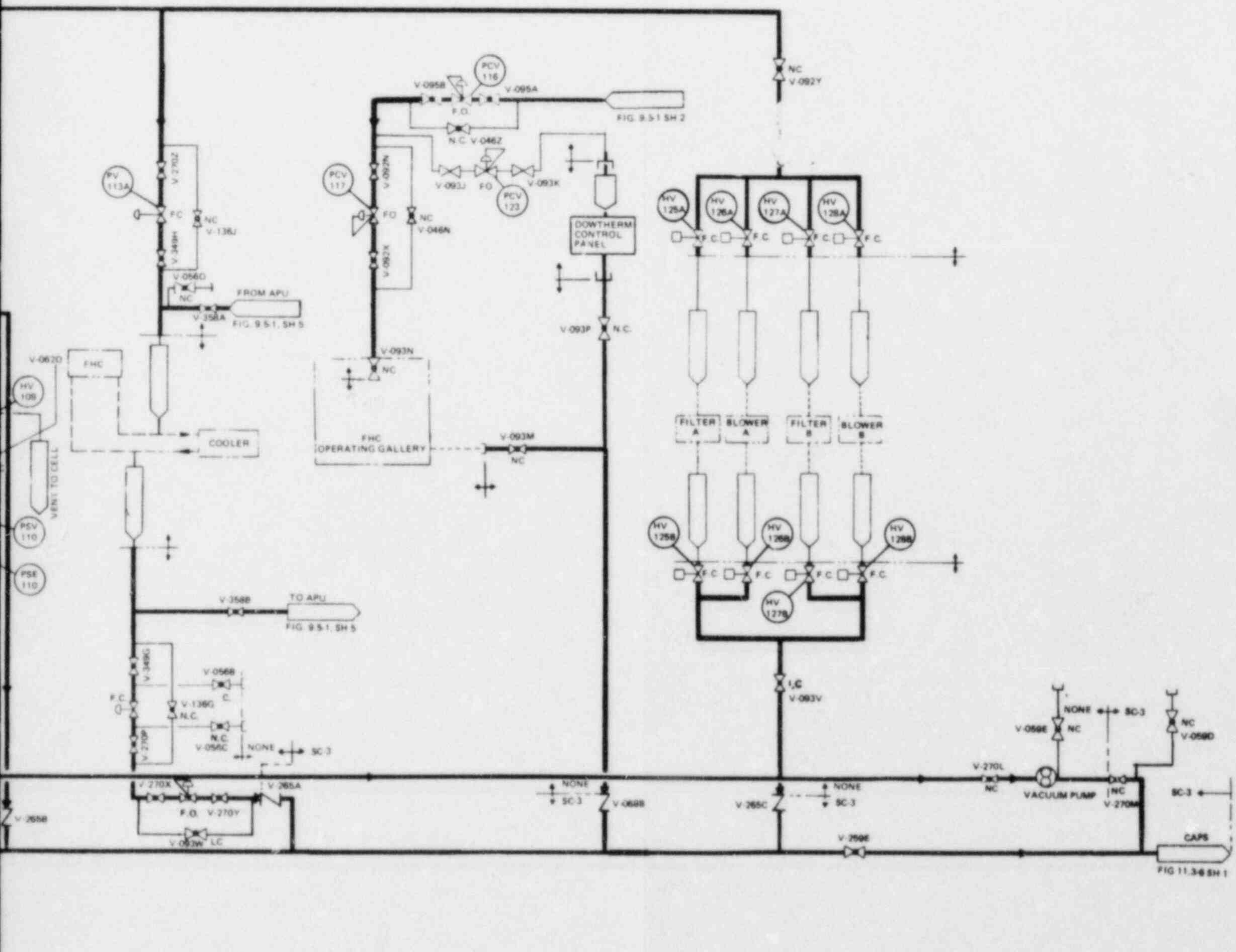

Figure 9.5-1 Argon Distribution In The Reactor Service Building (Sheet 3 of 5)

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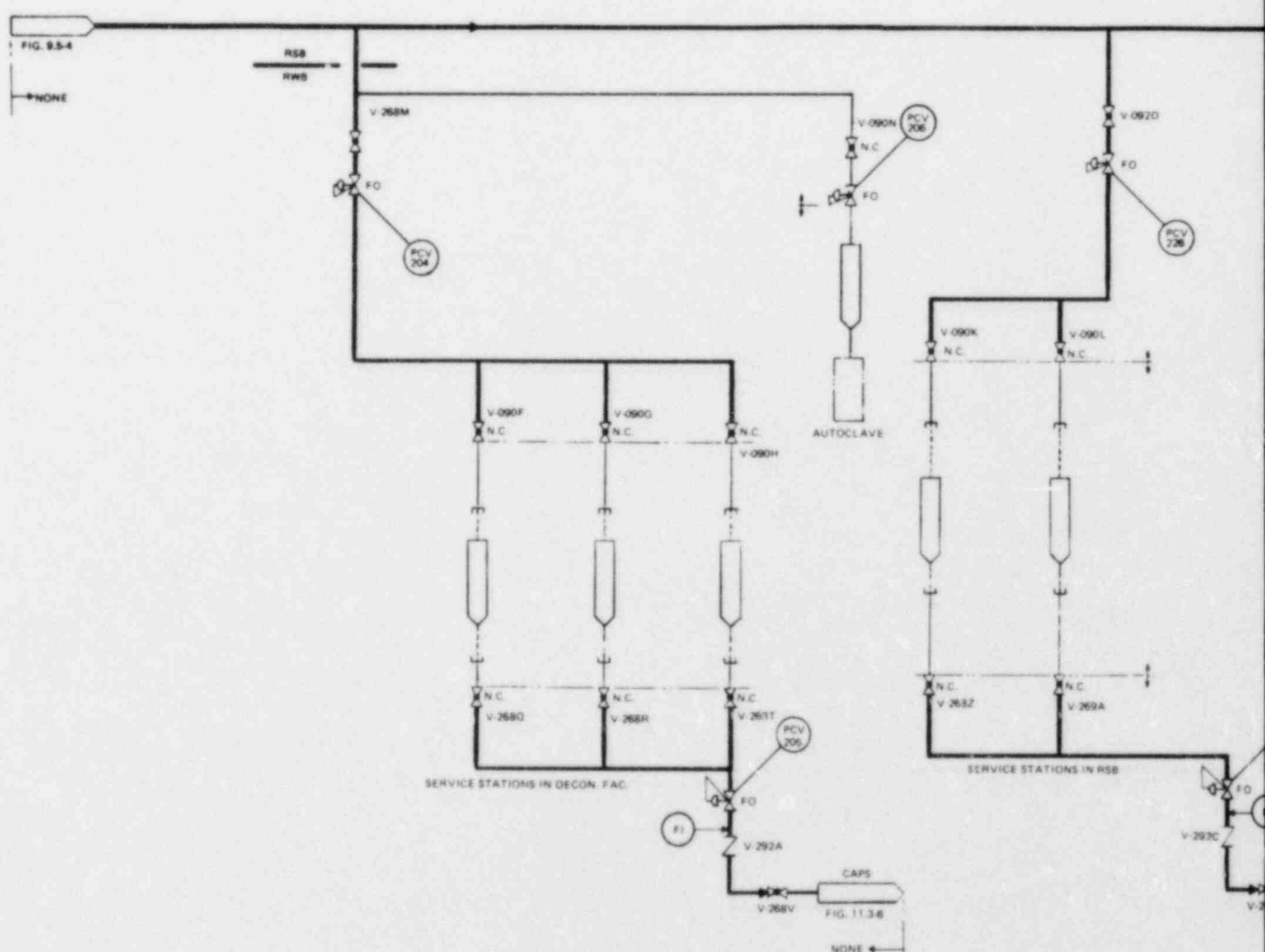


FIG. 9.5-4
NONE
RSB
RWB
V-268M
FO
PCV 204
V-090N
PCV 206
N.C.
FO
AUTOCLAVE
V-092O
FO
PCV 226
V-090K
N.C.
V-090L
N.C.
V-090F
N.C.
V-090G
N.C.
N.C.
V-090H
N.C.
V-268Q
N.C.
V-268R
N.C.
V-268T
PCV 205
N.C.
V-268Z
N.C.
V-269A
SERVICE STATIONS IN DECON. FAC.
SERVICE STATIONS IN RSB
FO
FI
V-292A
V-293C
FO
CAPS
V-268V
FIG. 11.3-6
NONE

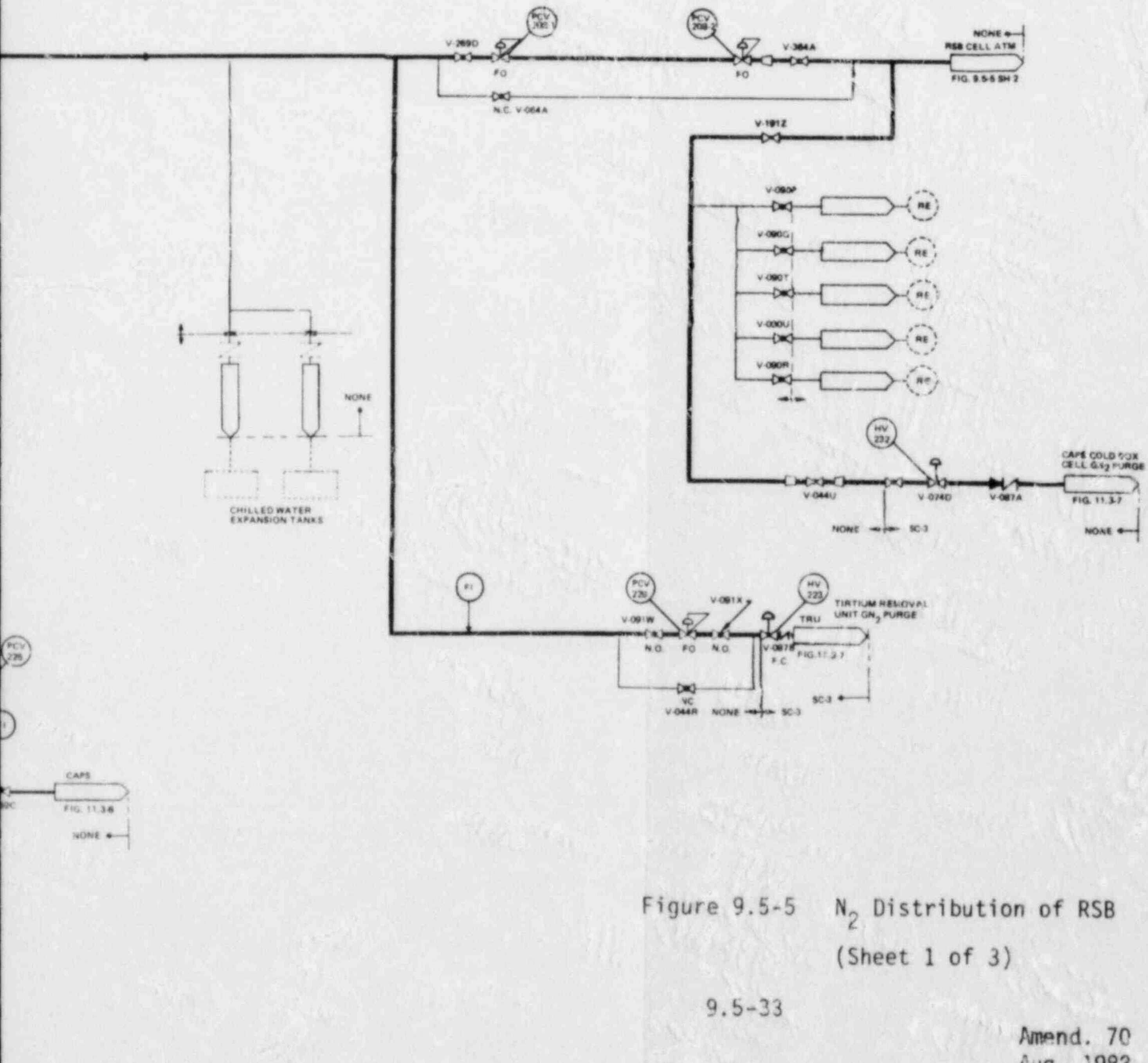
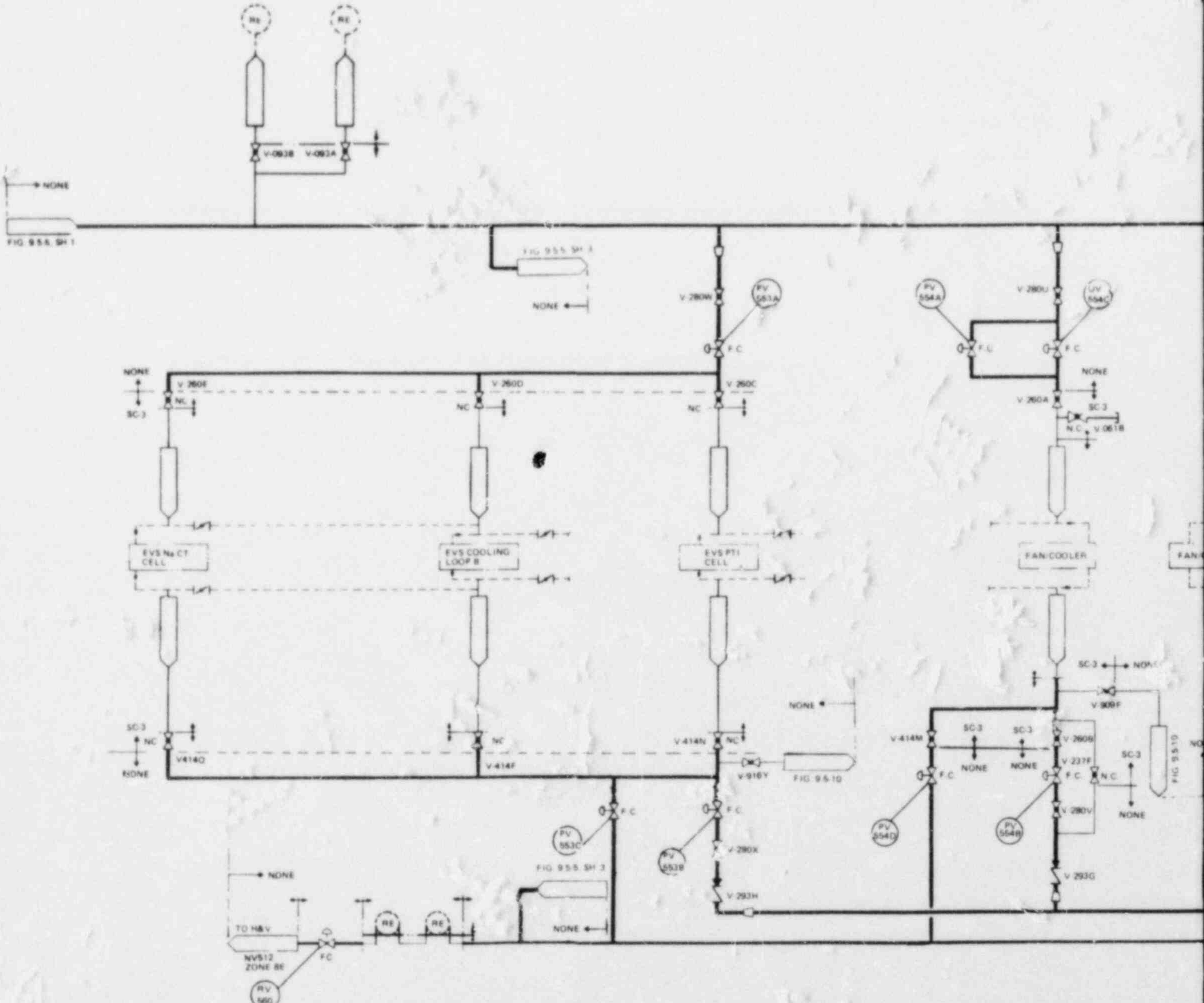


Figure 9.5-5 N_2 Distribution of RSB
(Sheet 1 of 3)



RE
RE
V-093B
V-093A
NONE
FIG 9.5.5, SH 1
FIG 9.5.5, SH 3
NONE
V-280W
PV 553A
F.C.
PV 554A
V-280U
UV 554C
F.U
F.C
NONE
V-260E
NL
SC-3
V-260D
NC
V-260C
NC
V-260A
SC-3
N.C.
V-061B
EVS Na CT CELL
EVS COOLING LOOP B
EVS PTI CELL
FAN/COOLER
SC-3
V-909F
NONE
SC-3
NC
V414Q
NONE
NC
V-414F
V-414N
NC
V-916Y
FIG 9.5.10
V-414M
SC-3
NONE
SC-3
NONE
V-260B
V-237F
F.C.
N.C.
SC-3
NONE
F.C
PV 554D
PV 554B
V-280V
FIG 9.5.10
F.C
PV 553C
F.C
PV 553B
V-280X
V-293H
V-293G
FIG 9.5.5, SH 3
NONE
NONE
TO H&V
NV512
ZONE 8E
F.C
RE
RE
RV 560

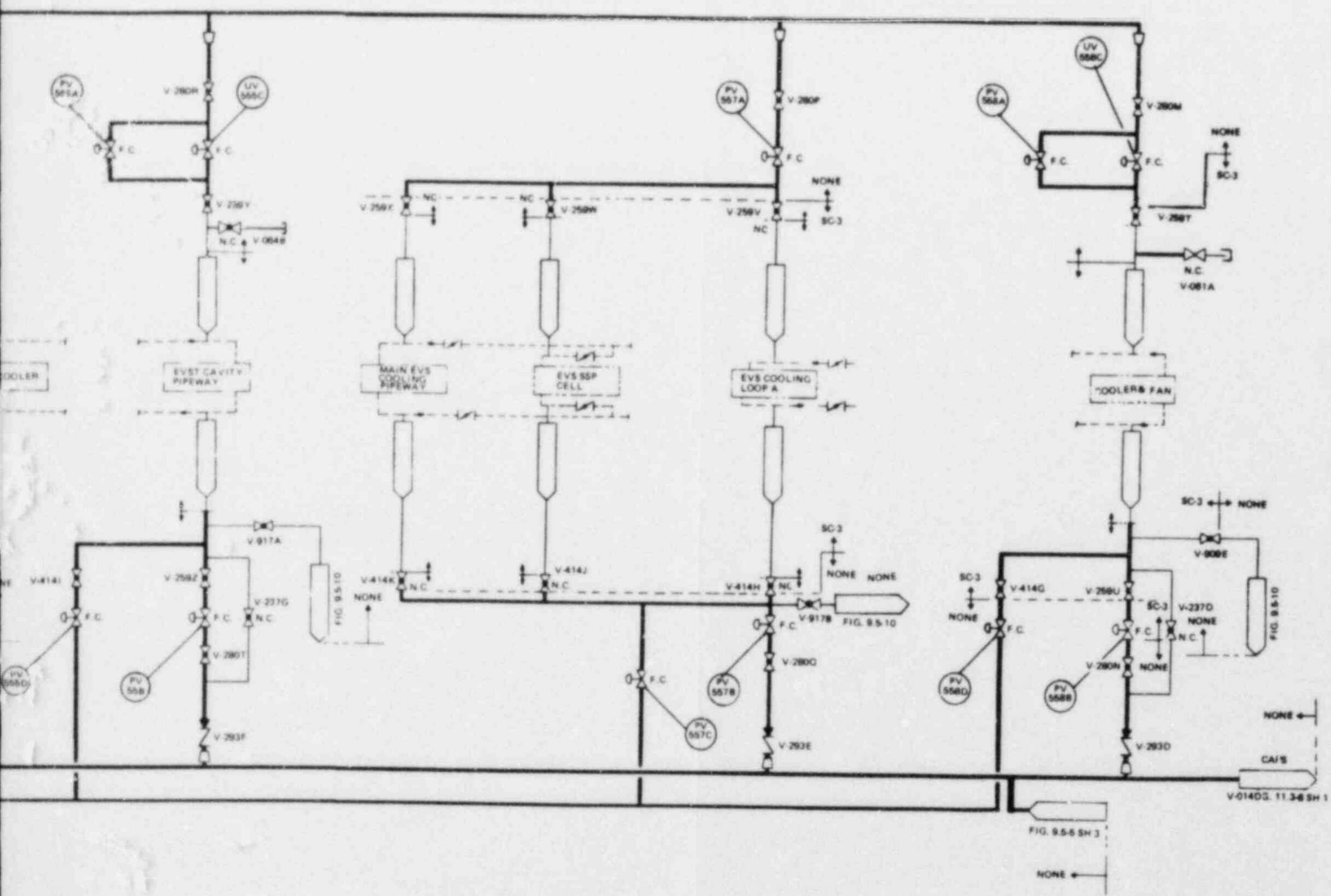
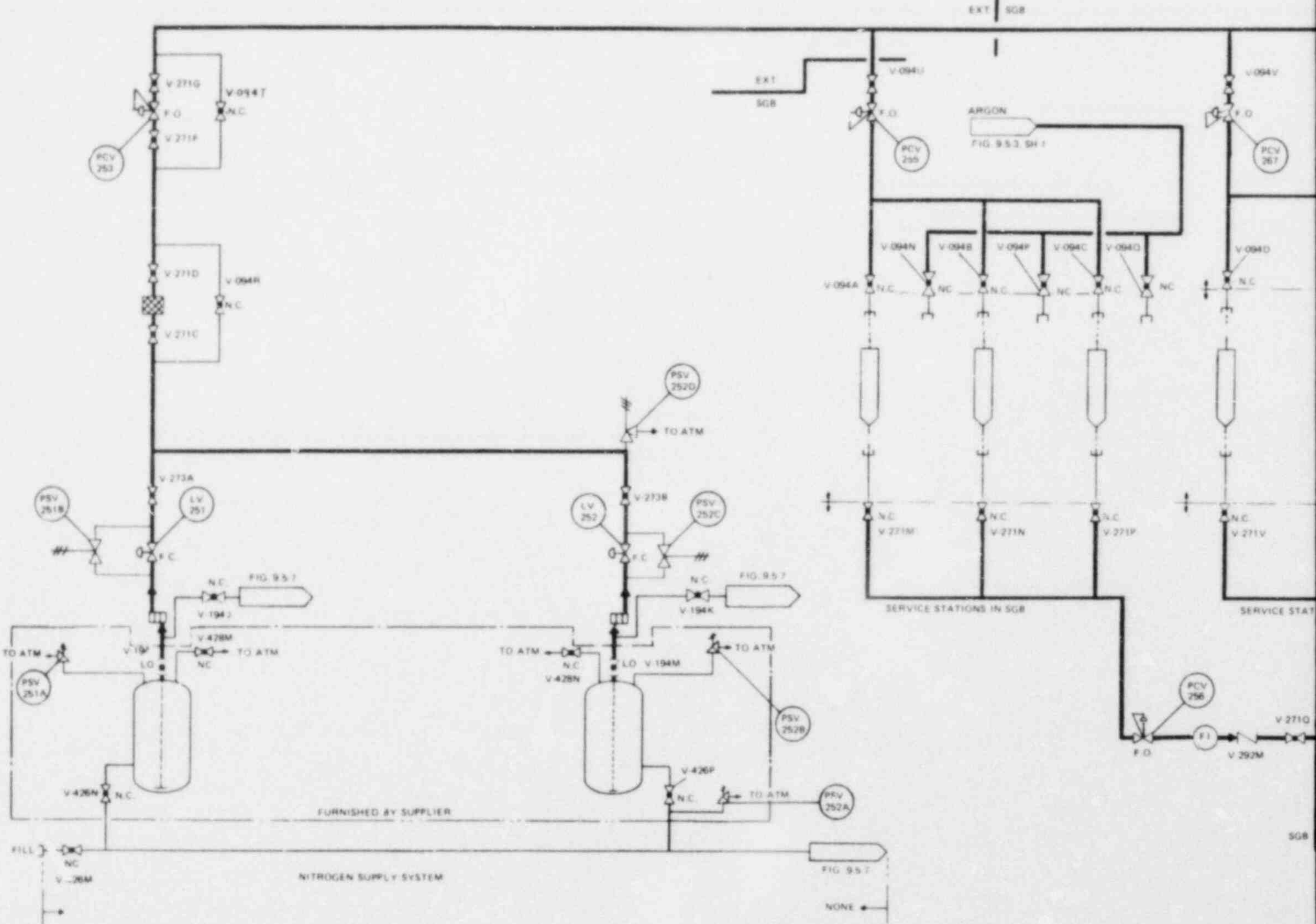


Figure 9.5-5 N_2 Distribution of RSB

(Sheet 2 of 3)


EXT
SGB
ARGON
FIG. 9.5.3, SH 1
PCV 253
PCV 255
PCV 257
PCV 256
V-271G
V-271F
V-271D
V-271C
V-094R
V-094U
V-094V
V-094A
V-094N
V-094B
V-094F
V-094C
V-094Q
V-094D
F.O.
F.C.
N.C.
NC
PSV 252D
TO ATM
V-273A
V-273B
PSV 251B
LV 251
LV 252
PSV 252C
V-271M
V-271N
V-271P
V-271V
SERVICE STATIONS IN SGB
FIG. 9.5.7
V-194J
V-194K
V-428M
V-428N
LO
V-194M
PSV 251A
PSV 252B
PSV 252A
V-426N
V-426F
FURNISHED BY SUPPLIER
FILL
NITROGEN SUPPLY SYSTEM
NONE
FI
V-292M
V-271Q
SGB

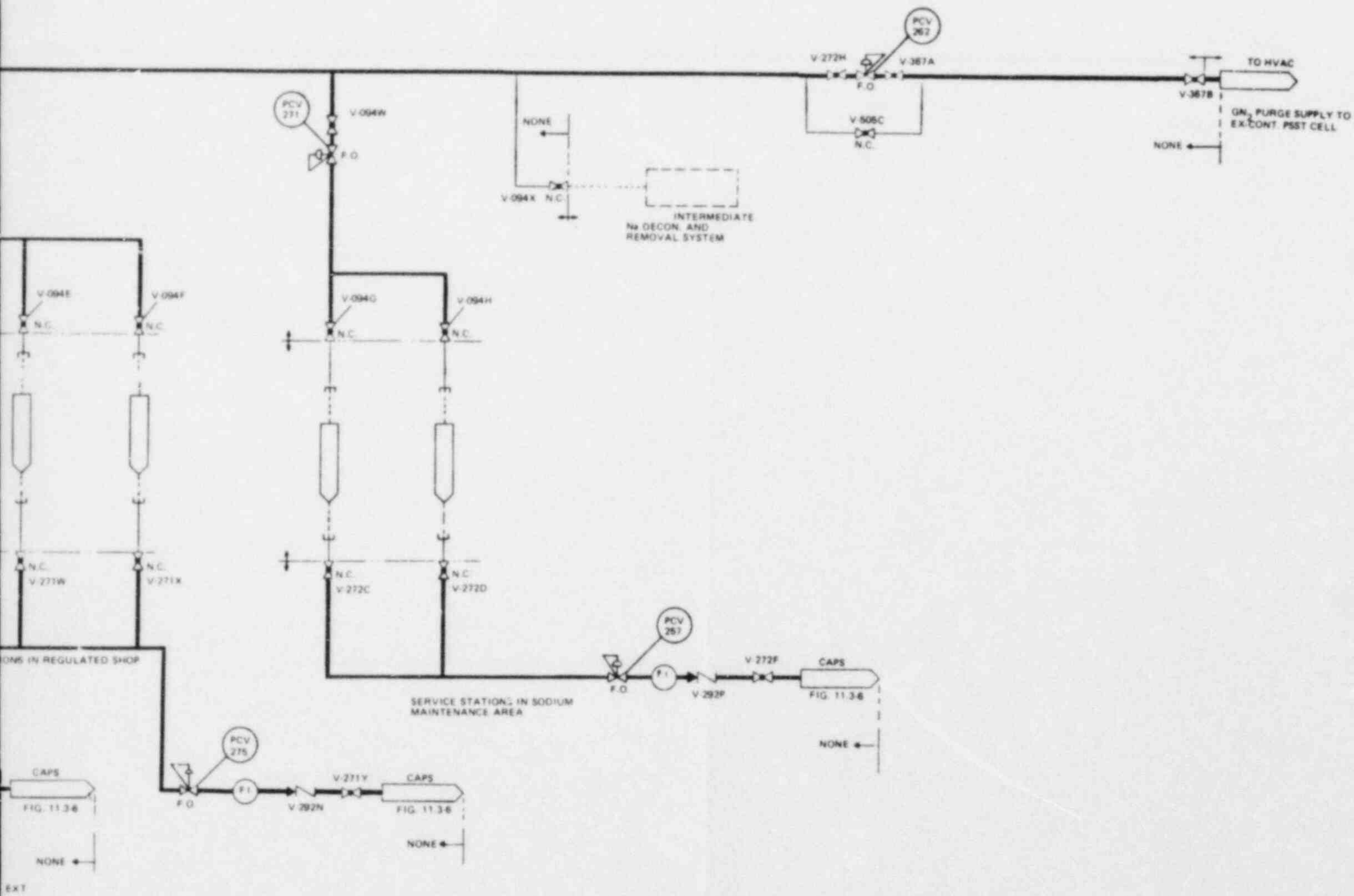
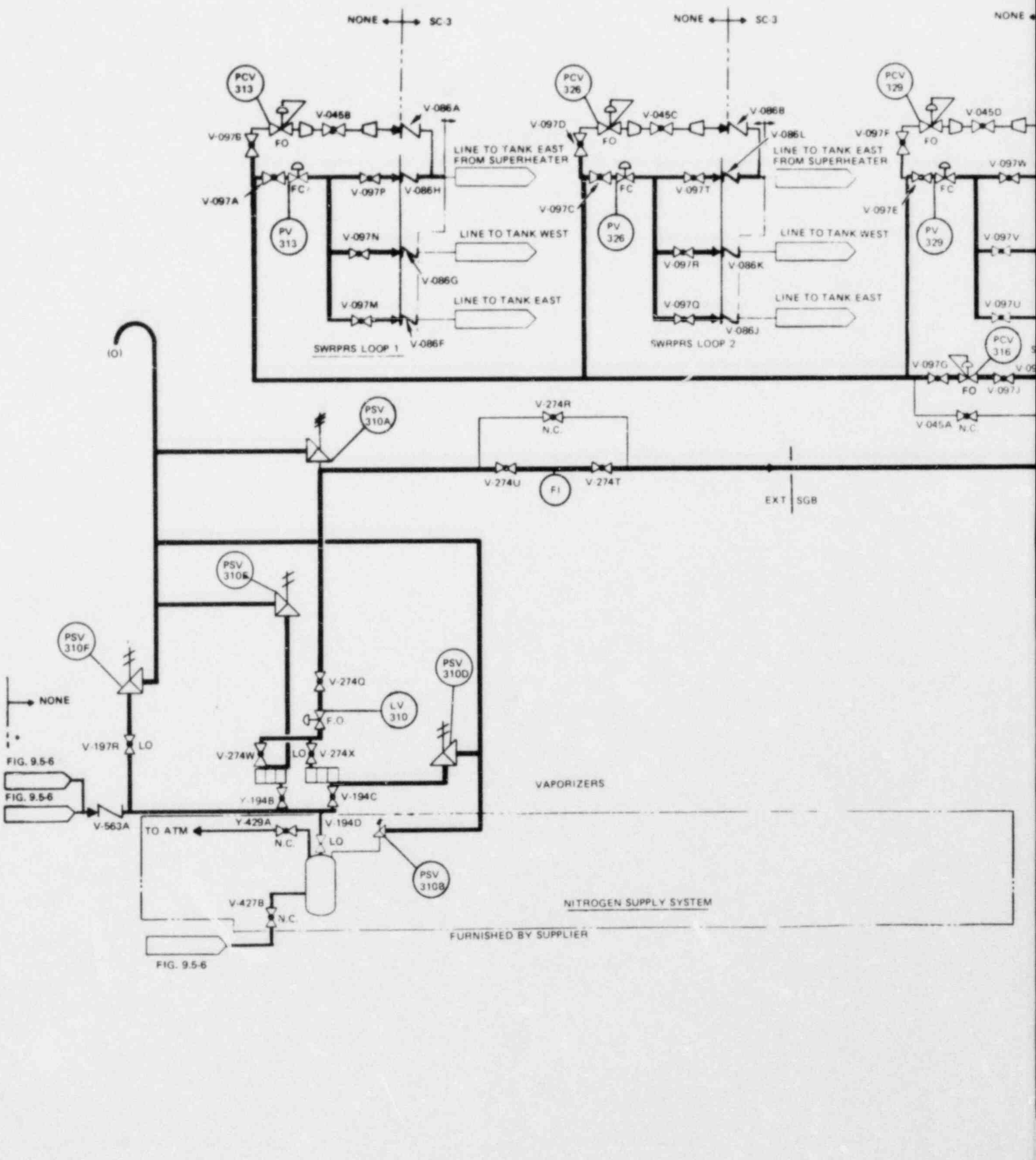


Figure 9.5-6
N_2 Distribution in SGB



NONE
SC-3
NONE
SC-3
NONE
PCV 313
V-097B
FO
V-045B
V-086A
LINE TO TANK EAST FROM SUPERHEATER
V-097A
FC
V-097P
V-086H
PV 313
V-097N
LINE TO TANK WEST
V-086G
V-097M
LINE TO TANK EAST
V-086F
SWRPRS LOOP 1
PCV 326
V-097D
FO
V-045C
V-086B
V-086L
LINE TO TANK EAST FROM SUPERHEATER
V-097C
FC
V-097T
PV 326
V-097R
V-086K
LINE TO TANK WEST
V-097Q
LINE TO TANK EAST
V-086J
SWRPRS LOOP 2
PCV 329
V-097F
FO
V-045D
V-097W
V-097E
FC
PV 329
V-097V
V-097U
PCV 316
V-097G
FO
V-097J
V-045A
N.C.
(O)
PSV 310A
V-274R
N.C.
V-274U
FI
V-274T
EXT
SGB
PSV 310E
PSV 310F
NONE
PSV 310D
V-274Q
LV 310
F.O.
V-197R
LO
FIG. 9.5-6
FIG. 9.5-6
V-274W
LO
V-274X
V-194B
V-194C
VAPORIZERS
V-563A
TO ATM
V-429A
N.C.
V-194D
LO
PSV 310B
V-427B
N.C.
NITROGEN SUPPLY SYSTEM
FURNISHED BY SUPPLIER
FIG. 9.5-6

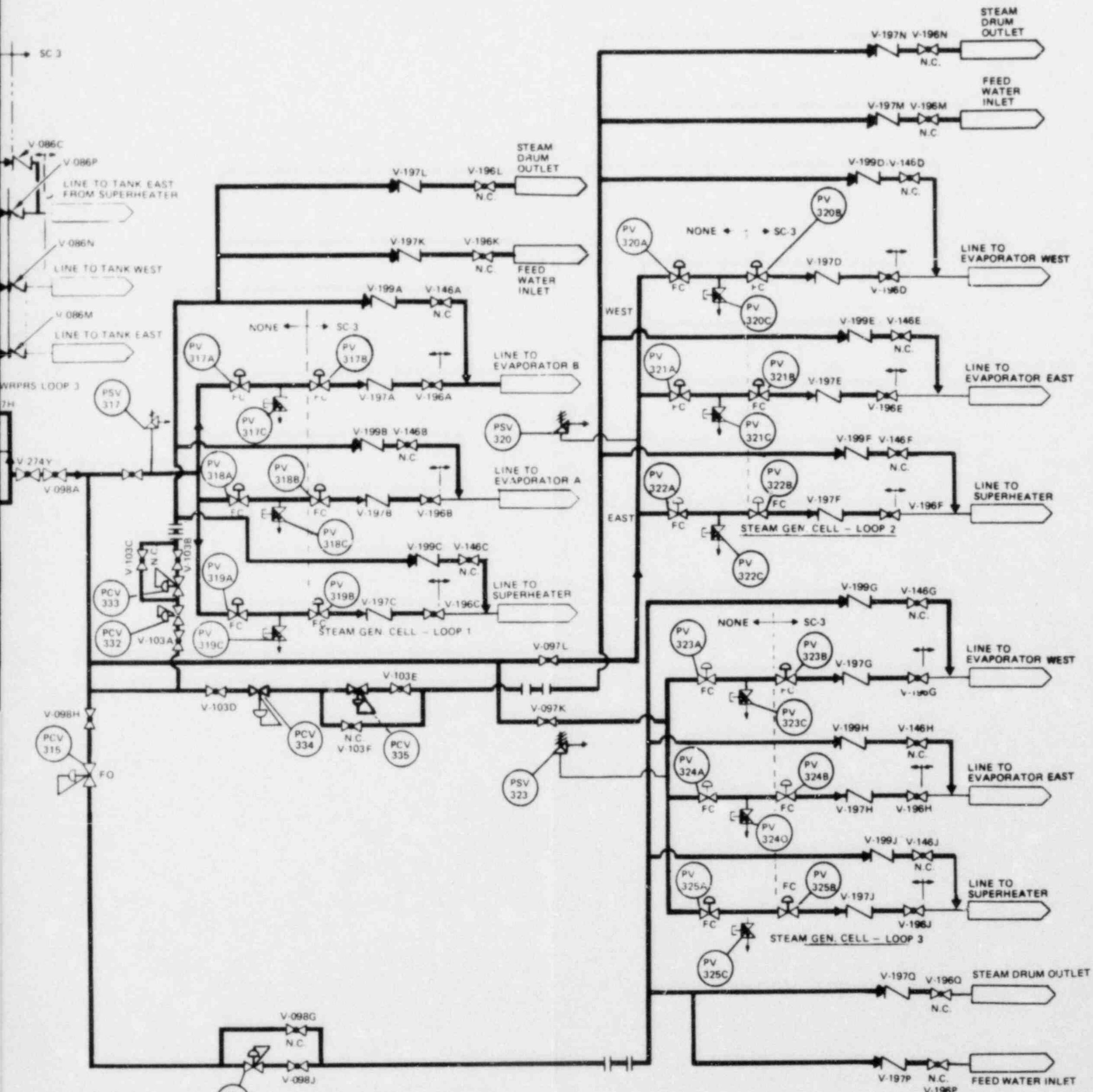
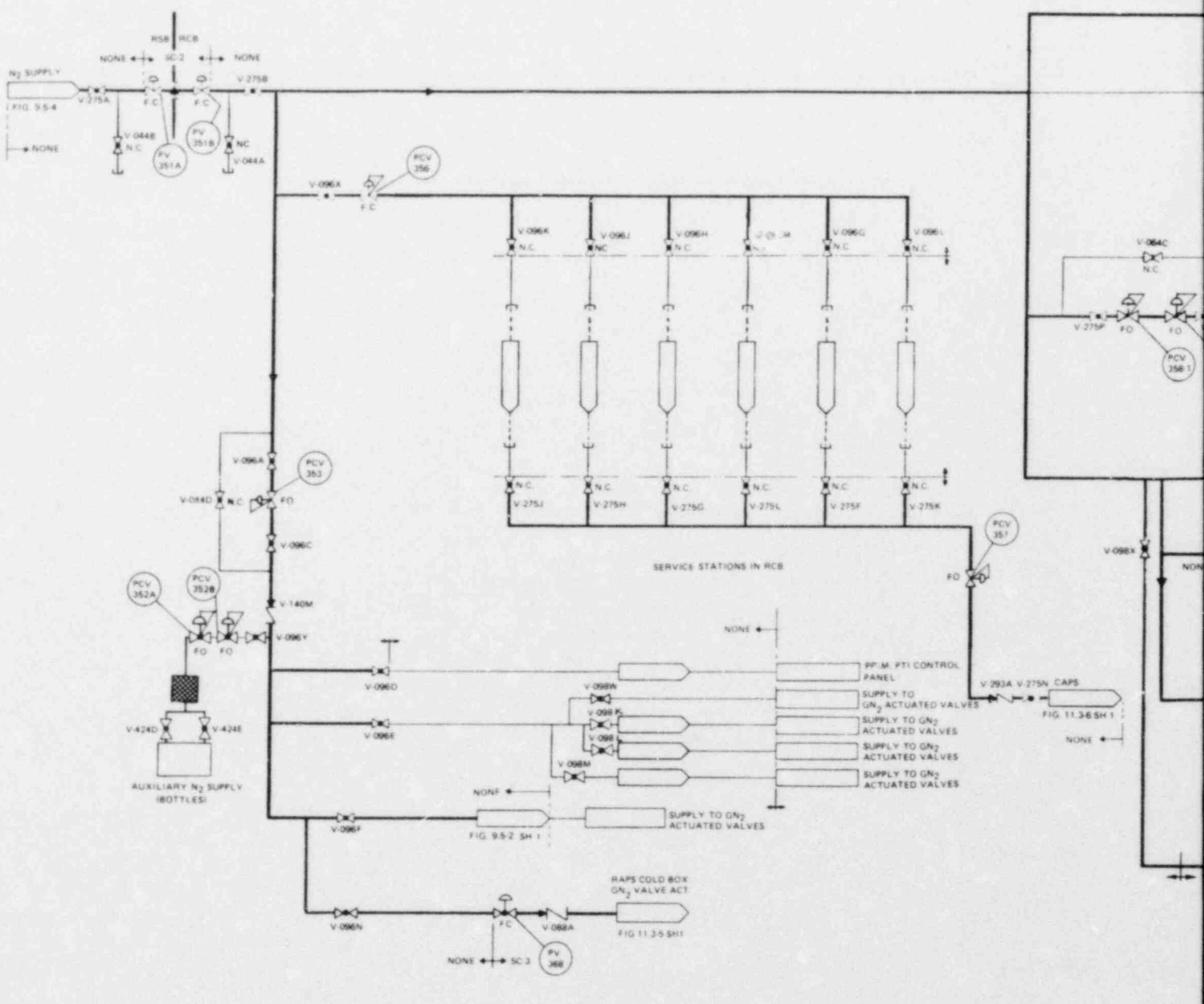


FIGURE 9.5-7 N_2 SUPPLY, SGB Na REACTION



RSB
RCB
SC-2
NONE
N_2 SUPPLY
FIG. 9.5-4
V-275A
V-275B
V-044B
V-044A
PV 351A
PV 351B
F.C.
N.C.
V-096X
PCV 356
V-096K
V-096J
V-096H
V-096G
V-096L
V-275J
V-275H
V-275G
V-275L
V-275F
V-275K
SERVICE STATIONS IN RCB
V-096A
PCV 353
V-044D
FO
V-096C
V-140M
V-096Y
PCV 352A
PCV 352B
V-424D
V-424E
AUXILIARY N_2 SUPPLY (BOTTLES)
V-096D
PPIM PTI CONTROL PANEL
V-098W
SUPPLY TO GN_2 ACTUATED VALVES
V-096E
V-098M
V-096F
FIG. 9.5-2 SH 1
V-096N
V-082A
FC
PV 368
SC-3
RAPS COLD BOX GN_2 VALVE ACT
FIG 11.3-5 SH1
PCV 357
V-293A
V-275N
CAPS
FIG. 11.3-6 SH 1
V-084C
V-275P
PCV 358
V-098X

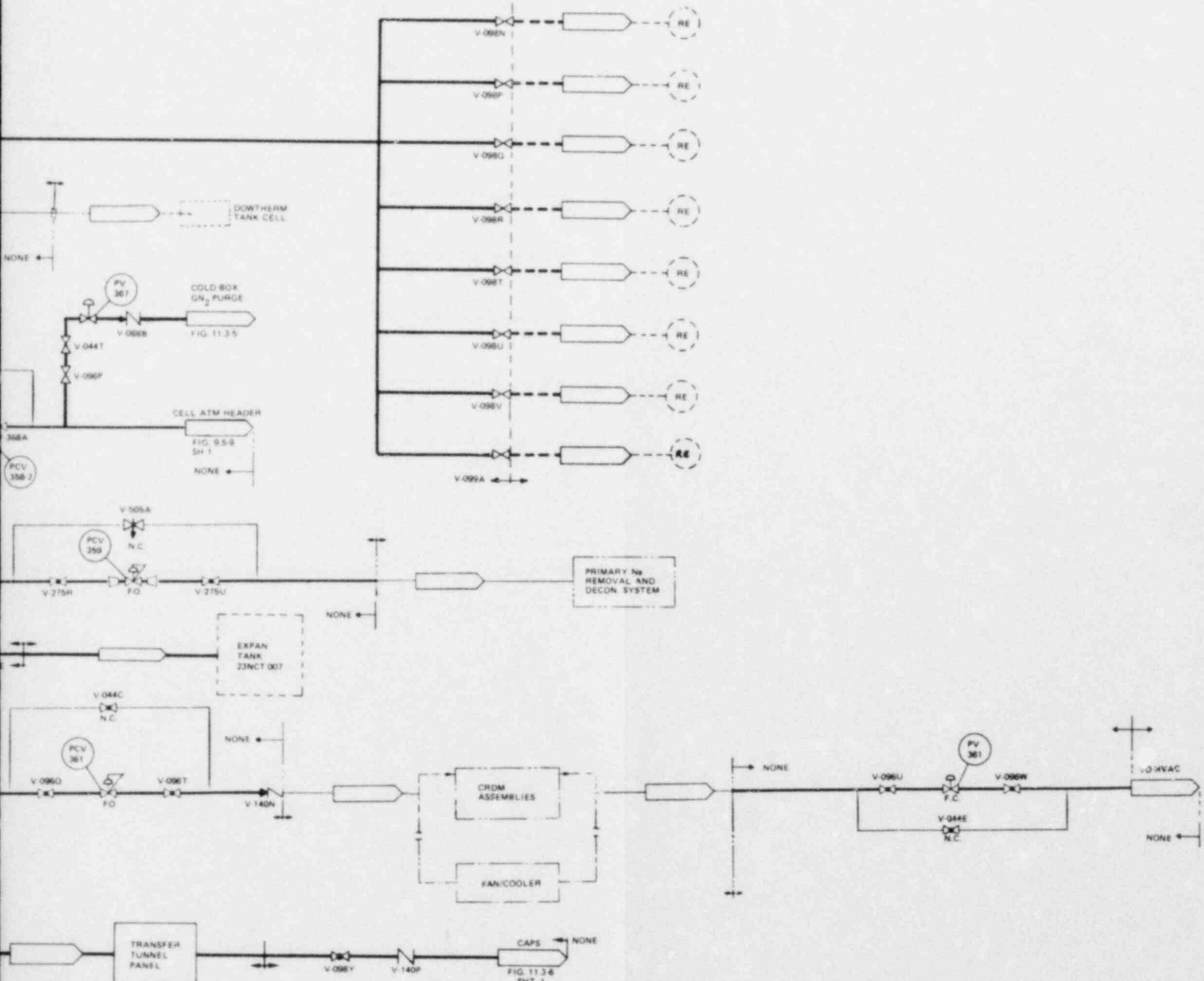
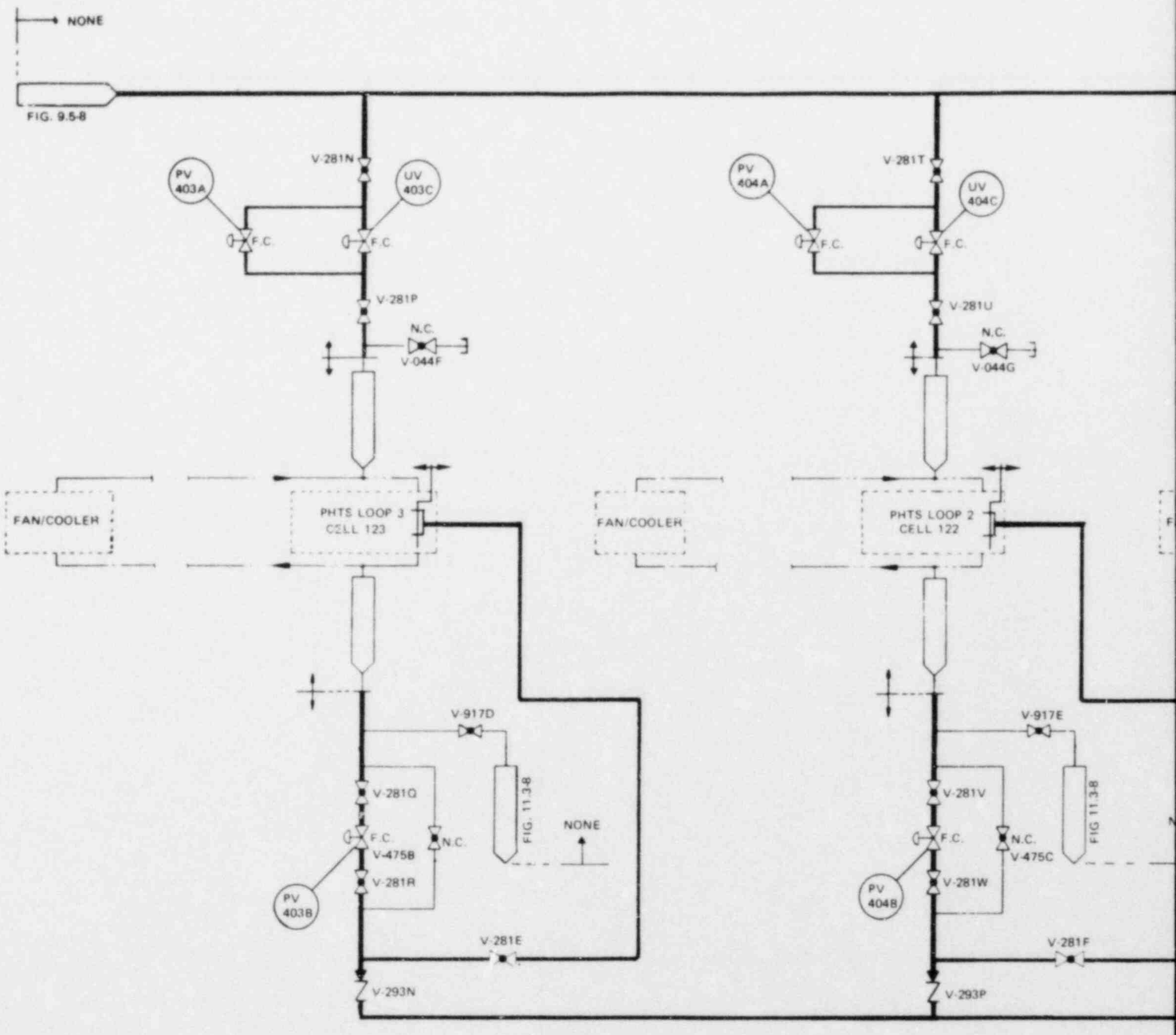


Figure 9.5-8
RCB N_2 Distribution High Pressure



NONE
FIG. 9.5-8
V-281N
PV 403A
UV 403C
F.C.
F.C.
V-281P
N.C.
V-044F
FAN/COOLER
PHTS LOOP 3
CELL 123
V-917D
V-281Q
F.C.
V-475B
N.C.
V-281R
PV 403B
FIG. 11.3-8
NONE
V-281E
V-293N
V-281T
PV 404A
UV 404C
F.C.
F.C.
V-281U
N.C.
V-044G
FAN/COOLER
PHTS LOOP 2
CELL 122
V-917E
V-281V
F.C.
N.C.
V-475C
V-281W
PV 404B
FIG 11.3-8
V-281F
V-293P

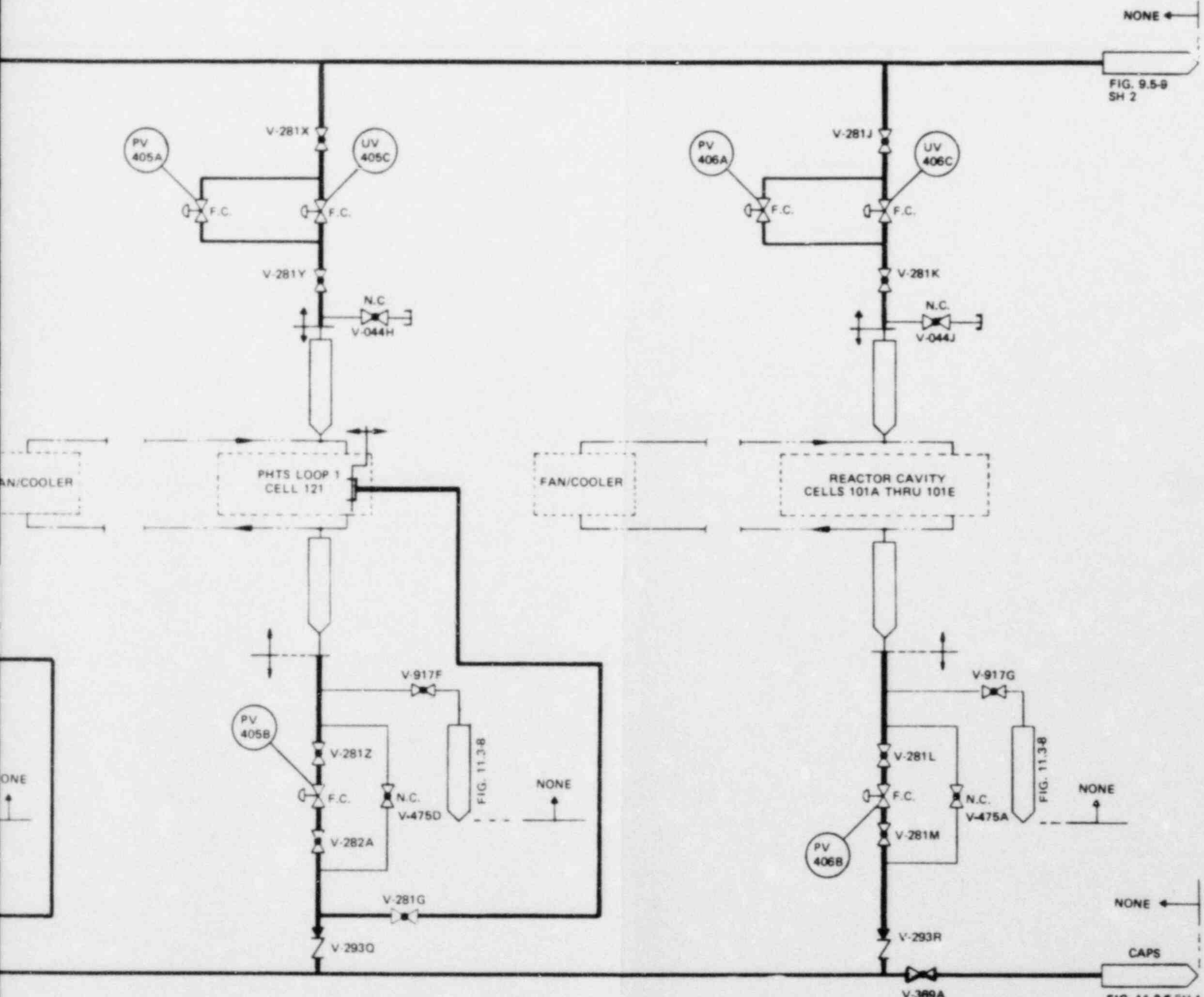
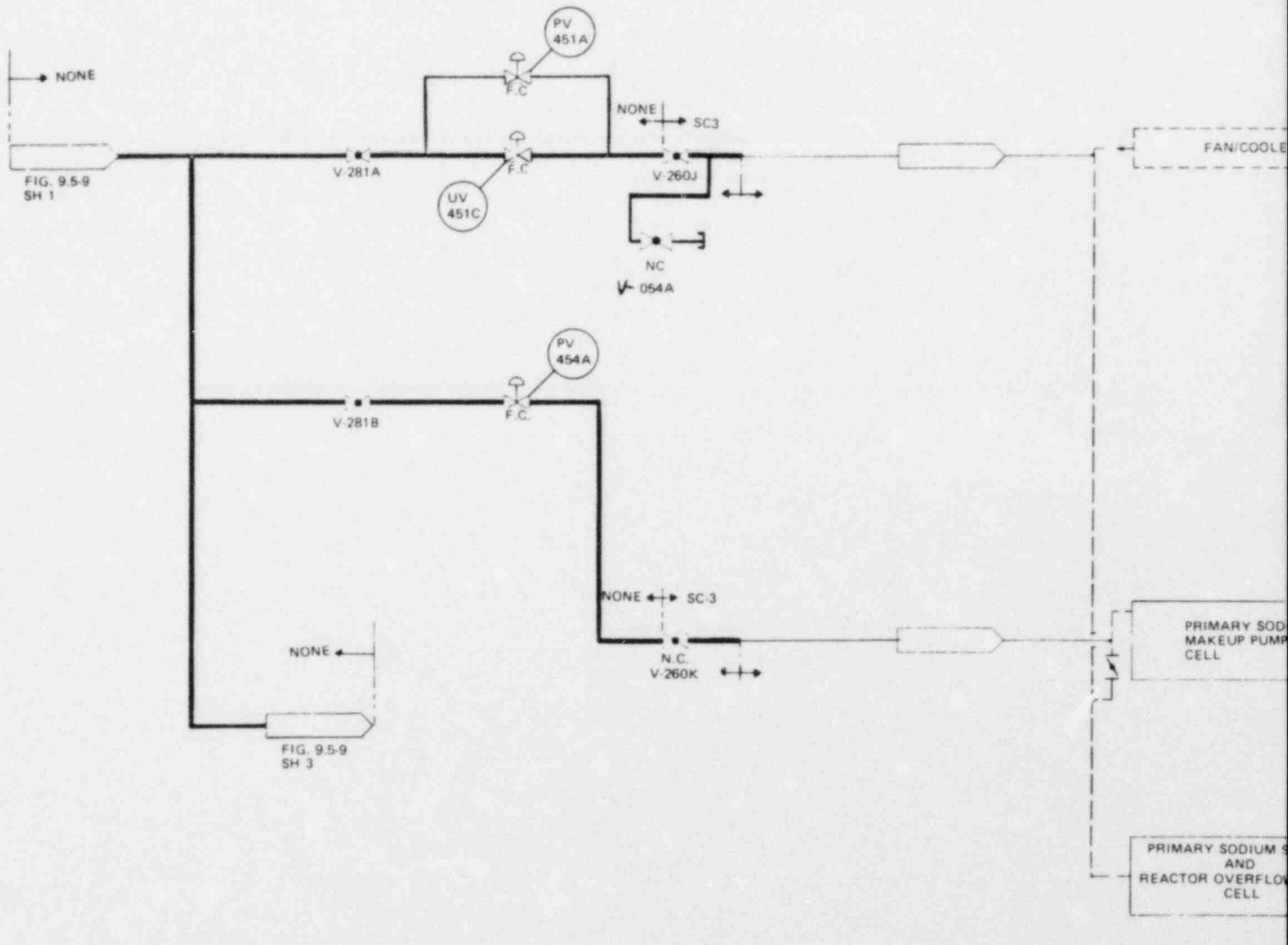


Figure 9.5-9 N_2 Distribution RCB, Low Pressure

(Sheet 1 of 4)


NONE
PV
451A
F.C
NONE
SC3
F.C
V-281A
V-260J
FIG. 9.5-9
SH 1
UV
451C
NC
V-054A
FAN/COOLE
PV
454A
F.C.
V-281B
NONE
SC-3
NONE
N.C.
V-260K
FIG. 9.5-9
SH 3
PRIMARY SOD
MAKEUP PUMP
CELL
PRIMARY SODIUM S
AND
REACTOR OVERFLOW
CELL

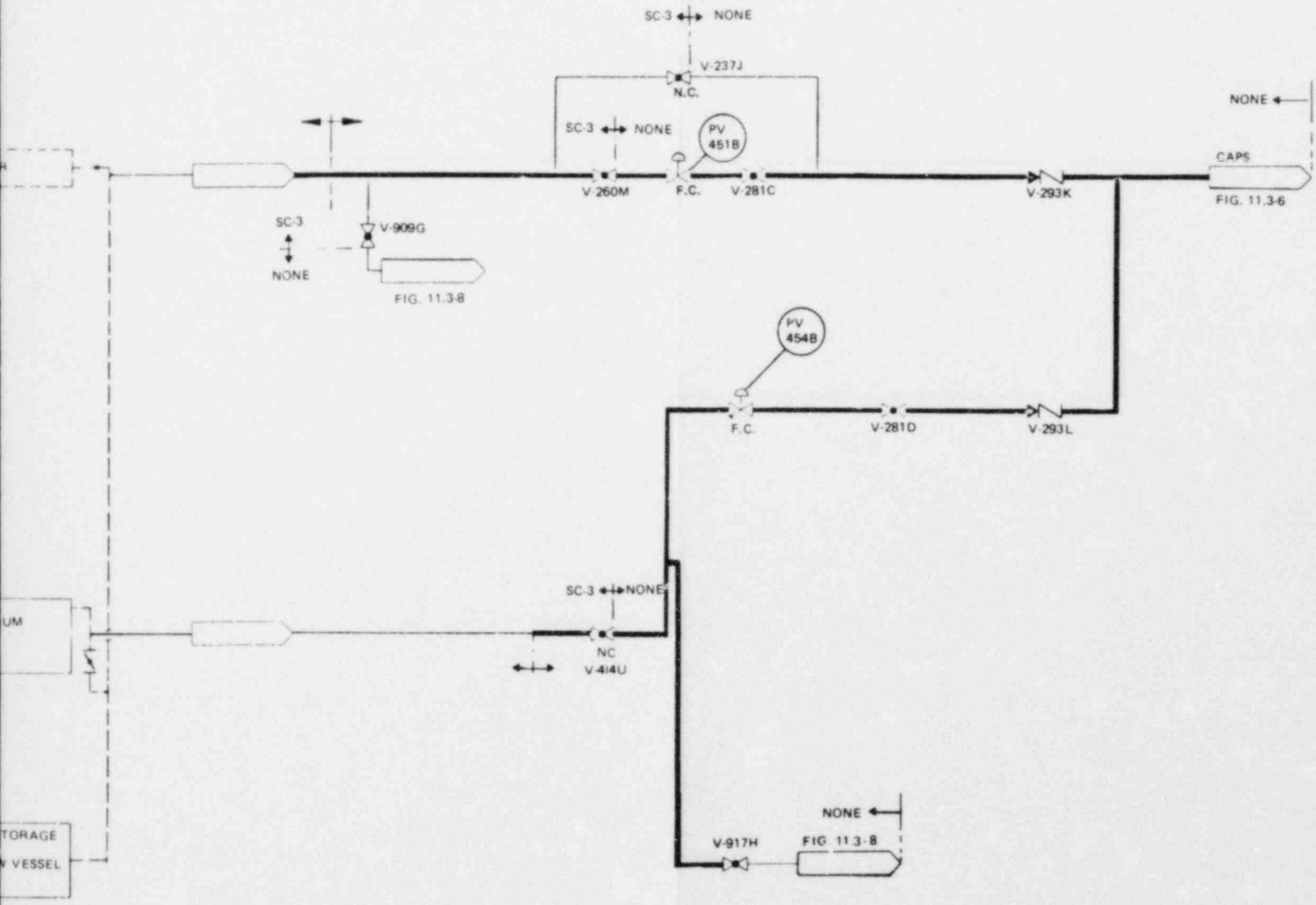


Figure 9.5-9 N_2 Distribution RCB, Low Pressure

(Sheet 2 of 4)

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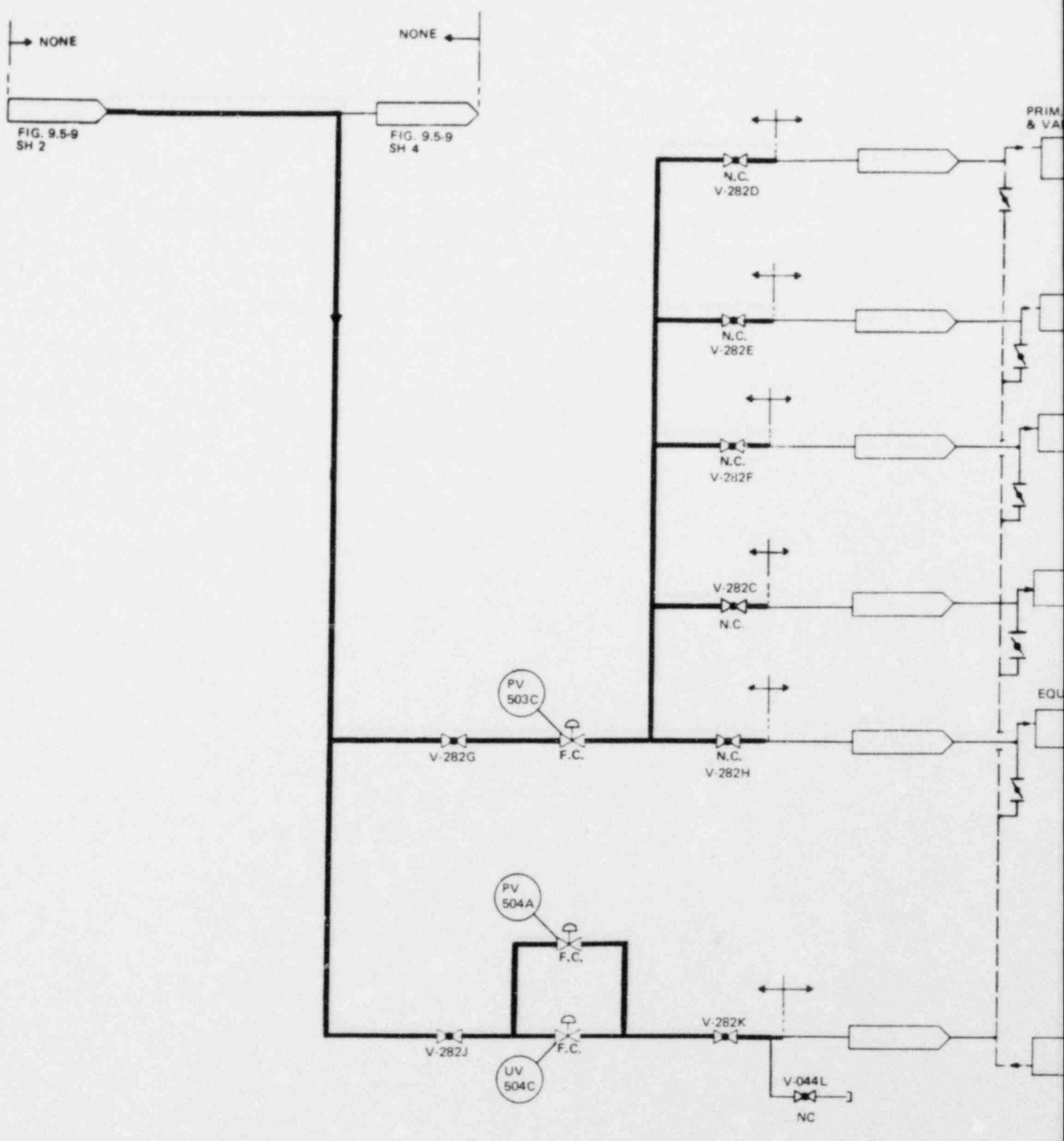


NONE
NONE
FIG. 9.5-9
SH 2
FIG. 9.5-9
SH 4
PRIM.
& VA
N.C.
V-282D
N.C.
V-282E
N.C.
V-282F
V-282C
N.C.
EQU
PV
503C
V-282G
F.C.
N.C.
V-282H
PV
504A
F.C.
V-282J
F.C.
UV
504C
V-282K
V-044L
NC

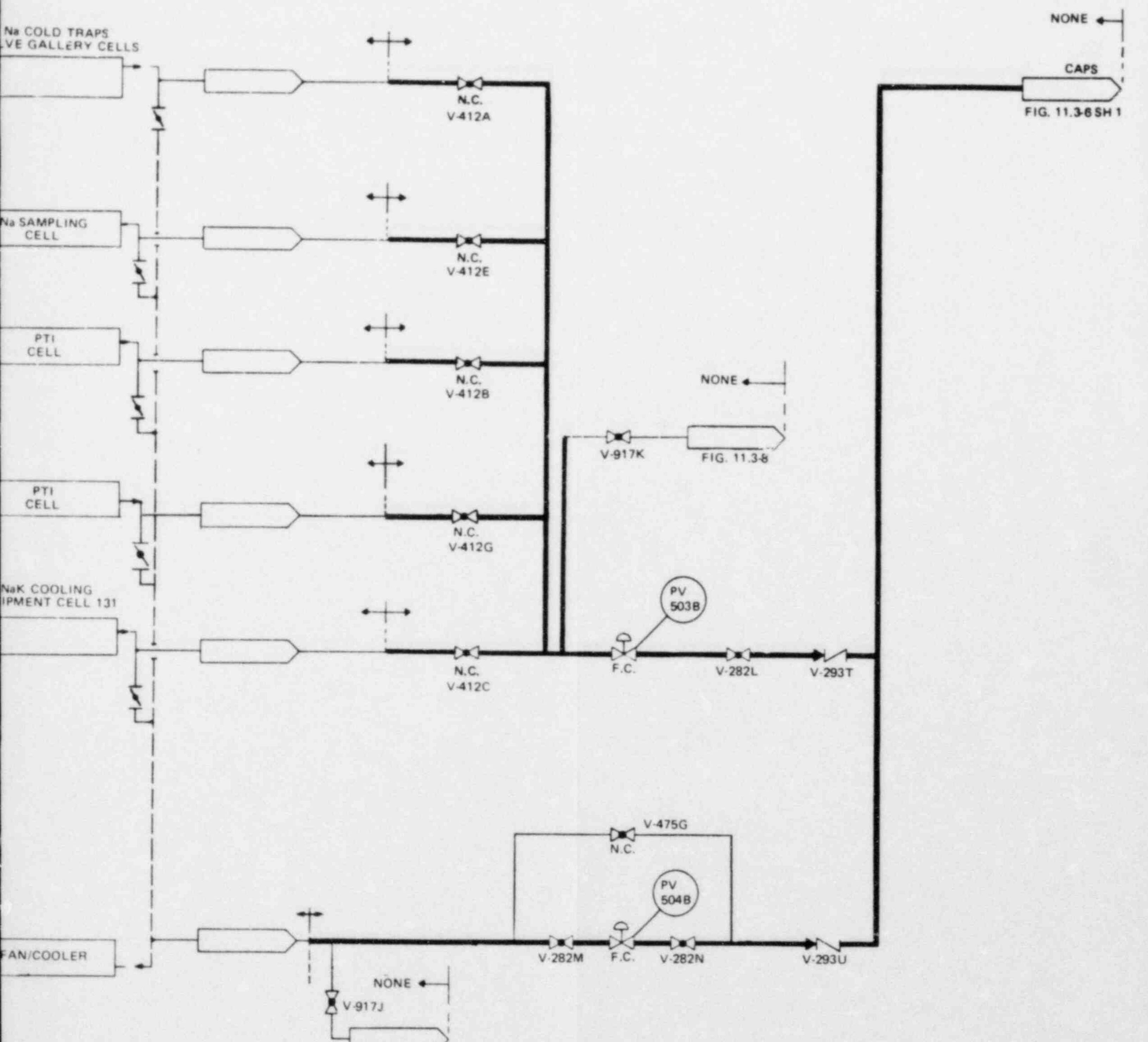
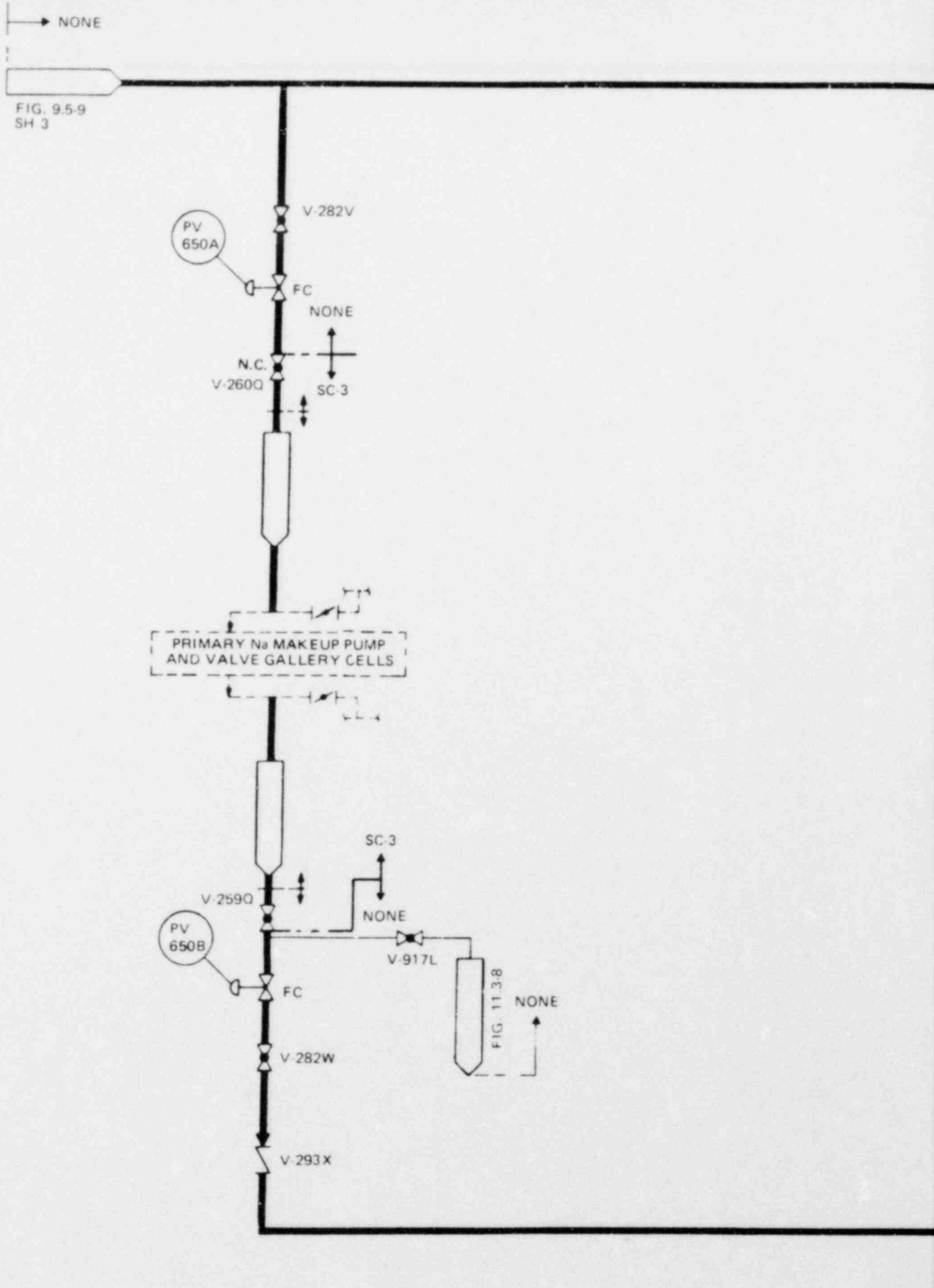


Figure 9.5-9 N_2 Distribution RCB, Low Pressure

(Sheet 3 of 4)


NONE
FIG. 9.5-9
SH 3
V-282V
PV
650A
FC
NONE
N.C.
V-260Q
SC-3
PRIMARY Na MAKEUP PUMP
AND VALVE GALLERY CELLS
SC-3
V-259Q
NONE
PV
650B
V-917L
FC
FIG. 11.3-8
NONE
V-282W
V-293X

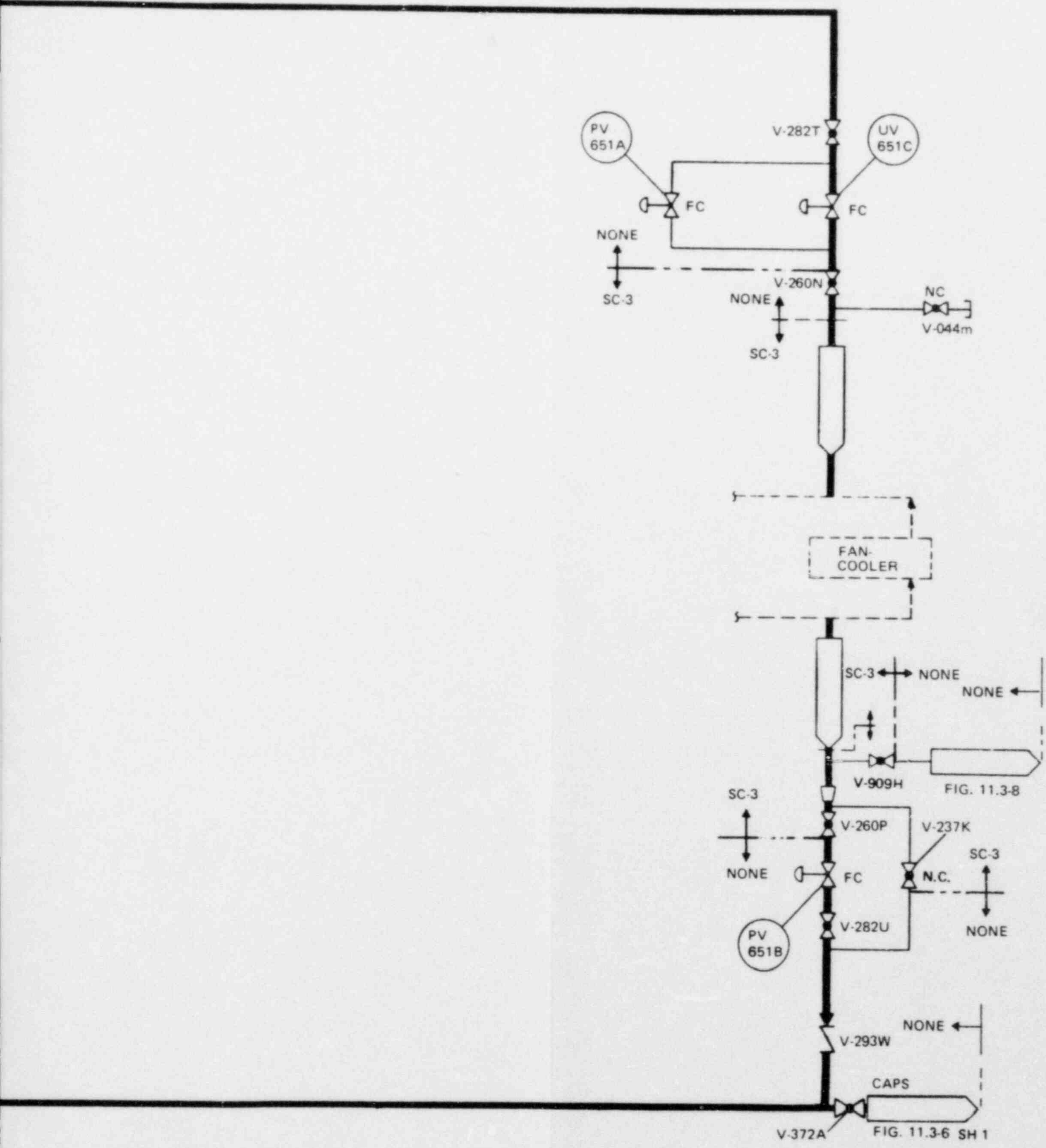


Figure 9.5-9 N_2 Distribution RCB, Low Pressure

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TABLE 11.3-2

GASEOUS RADIONUCLIDE CONCENTRATION IN
REACTOR COVER GAS*

Isotope	Inventory (Ci)	Concentration μCi/scc
Xe^{131m}	8.6	0.74
Xe^{133m}	2.8E+2	24.
Xe^{133}	5.0E+3	4.3E+2
Xe^{135m}	1.2E+3	1.1E+2
Xe^{135}	2.3E+4	1.9E+3
Xe^{138}	2.0E+3	1.8E+2
Kr^{83m}	7.5E+2	64
Kr^{85m}	1.8E+3	1.5E+2
Kr^{85}	0.16	1.4E-2
Kr^{87}	2.0E+3	1.7E+2
Kr^{88}	3.4E+3	2.9E+2
Ar^{39}	9.09**	0.783**
Ar^{41}	24.	2.0
Ne^{23}	8.9E+5	7.7E+4
H^3	1.7E-4	1.5E-5

*For the design condition
**After 30 years' operation

TABLE 11.3-3

ACTIVITY INVENTORIES IN RAPS PROCESS VESSELS

	RAPS Vacuum Vessel		RAPS Surge Vessel		RAPS Cryostill		Recycle Argon Vessels	
Isotope	Design (Ci)	Expected (Ci)	Design (Ci)	Expected (Ci)	Design (Ci)	Expected (Ci)	Design (Ci)	Expected (Ci)
Xe^{131m}	1.2	0.12	28	2.8	1.9E3	1.9E2	1.1E-3	1.1E-4
Xe^{133m}	38	3.8	8.2E2	82	1.1E4	1.1E3	3.1E-2	3.1E-3
Xe^{133}	6.9E2	69	1.6E4	1.6E3	4.7E5	4.7E4	0.61	6.1E-2
Xe^{135m}	24	2.4	32	3.2	2.0	0.20	6.6E-5	6.6E-6
Xe^{135}	2.5E3	2.5E2	4.0E4	4.0E3	8.8E4	8.8E3	1.1	0.11
Xe^{138}	35	3.5	44	4.4	2.5	0.25	8.2E-5	8.2E-6
Kr^{83m}	50	5.0	3.6E2	36	1.6E2	16	3.9E-3	3.9E-4
Kr^{85m}	1.7E2	17	2.0E3	2.0E2	2.1E3	2.1E2	3.8E-2	3.8E-3
Kr^{85}	2.2E-2	2.2E-3	0.52	5.2E-2	7.2E2	72	2.1E-5	2.1E-6
Kr^{87}	1.1E2	11	6.0E2	60	1.8E2	18	5.0E-3	5.0E-4
Kr^{88}	2.7E2	27	2.5E3	2.5E2	1.7E3	1.7E2	3.7E-2	3.7E-3
Ar^{39*}	3.5	3.5	81	81	28	28	49	49
Ar^{41}	1.8	1.8	13	13	2.5	2.5	2.2	2.2
Ne^{23}	17	17	0.97	0.97	3.4E-4	3.4E-4	1.3E-3	1.3E-3
H^3	6.7E-5	6.7E-5	1.6E-3	1.6E-3	5.4E-4	5.4E-4	9.5E-4	9.5E-4
Total	3.9E3	4.1E2	6.2E4	6.3E3	5.8E5	5.8E4	53	51

*After 30 years' operation

Fuel Handling Cell

The fuel handling cell (FHC) is designed to handle and inspect spent fuel in preparation for shipment off-site. The maximum gamma ray source term as a function of energy is shown in Table 12.1-27 for an expended assembly at four days after reactor shutdown. The maximum inherent neutron source is shown in Table 12.1-28 for three expended assemblies 80 days after reactor shutdown. These source strengths are based on the following design parameters:

a. The design assembly produces power, just before discharge, at 127 percent of the average assembly.

b. The fuel is being handled in the FHC at four days after reactor shutdown.

c. Each assembly has a volume of 1.173×10^4 cm^3 and an active height of 91.44 cm.

d. The energy spectrum is derived from the ENDF/B and RIBD code library as discussed previously.

The FHC shield design should accommodate the design assembly source given in Tables 12.1-27 and 12.1-28. The energy distribution of the inherent neutron source shown in Table 12.1-28 was treated as a Pu fast fission spectrum.

Fuel Handling Cell Argon Circulation System Sources

The FHC service cells have the following design basis:

A. FHC Argon Filter Cell

(1) Complete release into the FHC of all noble gas, halogen, and volatile fission products from the gaps and fission plenums of two failed fuel pins, is assumed. No credit is taken for iodine attenuation by sodium, since the pin failures are assumed to occur during handling of bare fuel assemblies.

(2) For conservatism in the filter shielding design, it is assumed that 100% of the released Cs and I are collected in the filters. For conservative cell shielding design it is assumed that only 60% of all Cs and I released is collected on the filters. The remaining 40% is equally distributed in the three support cells, and in the FHC itself. Since the filters are frequently removed, there will be no buildup of long-lived Cs-137.

(3) Activities are based on a decay time of 80 days.

(4) A FHC volume of 10,700 scf is assumed. Equipment (piping, filter banks, transitions) have void volume of 224 scf/cell.

(5) Cell specific activities are based on the radioactive releases (item 1) divided by the volume of the FHC.

Fuel handling cell, filter cell specific activity and FHC filter cell equipment activity appear in Table 12.1-29.

B. FHC Argon Blower Cell

Ten percent of the Cs and I activities released from the gaps and fission gas plenum of two failed pins are deposited on the inner surface of components. The long-lived Cs-137 activity has been multiplied by a factor of five to account for subsequent pin failures during plant life at a rate of one pin every three years.

FHC argon blower particulate activity is defined in Table 12.1-29.

Gas activity in the FHC Argon Blower Cell is the same as defined above and Table 12.1-29. Equipment in the FHC Argon Blower Cell (storage tank, grapple blowers, valves, piping) have a gas volume of 46 scf/cell.

C. FHC Fan/Cooler Unit

Sources in the FHC Fan/Cooler Unit are defined on the same bases as described for the Argon Blower Cell.

Particulate activity in the FHC Fan/Cooler Unit are the same as defined for the Argon Blower Cell.

Gas activity in the FHC Fan/Cooler Unit has the same specific activity as defined for the Argon Filter Cell.

Equipment in the FHC Argon Blower Unit cell cooler fan assembly has a gas volume of 200 scf.

D. FHC Argon Purification Unit

Sources in the FHC Argon Purification Unit are on the same bases as defined for the Argon Blower Cell.

Particulate activity in the FHC Argon Purification Unit is the same activity as defined for the Argon Blower Cell.

Gas activity in the FHC Argon Purification Unit is the same specific activity as defined for the Argon Blower Cell.

Equipment (Argon Purification Unit) in the cell has a gas volume
49 of 100 scf.

CHAPTER 13.0 CONDUCT OF OPERATIONS

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CHAPTER 13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF THE APPLICANT

Contract AT (49-18)-12 has been established to design, construct and operate a Liquid Metal Fast Breeder Reactor (LMFBR) demonstration plant. The parties to the contract are the Department of Energy (DOE), the Tennessee Valley Authority (TVA), the Commonwealth Edison Company (CE), and the Project Management Corporation (PMC). The organizational structure of the applicant (DOE, PMC, and TVA) is covered in Section 1.4.

TVA, as part of its lead role responsibility described in Section 1.4, will be responsible for the safe operation of the CRBRP.

13.1.1 Project Organization

13.1.1.1 Functions, Responsibilities, and Authorities of Project Participants

The functions, responsibilities, and authorities of Project participants are described in Sections 1.4 and 1.4.2. The qualification requirements of Project participants are described in Section 1.4.4.

13.1.1.2 Applicants' In-House Organization

This material is covered in Section 1.4.2.

13.1.1.3 Interrelationships with Contractors and Suppliers

This material is covered in Section 1.4.3.

13.1.1.4 Department of Energy (DOE) Participation

The participation of DOE in the Clinch River Breeder Reactor Plant (CRBRP) Project is described in Section 1.4. In addition, DOE participates in R&D in support of the CRBRP Project through its LMFBR base technology programs described in Section 1.5.

13.1.1.5 Technical Support for Operations

TVA's Office of Power will be responsible for carrying out the operator role for the Clinch River Breeder Reactor Plant. Within the TVA Office of Power, the Division of Nuclear Power (NUC PR) will be responsible for the operation and maintenance of the CRBRP. The TVA technical staff supporting the operation of the CRBRP will consist of the Nuclear Central Office (NCO) staff in Chattanooga and also support from other divisions and offices within TVA (Section 13.1.1.5.1). In addition, technical support will be supplied by the CRBRP Project Office, the Lead Reactor Manufacturer of Westinghouse Advanced Reactor Division (W-LRM),and by the AE (Burns and Roe) (Section 13.1.1.5.2).

13.1.1.5.1 TVA's Technical Staff

13.1.1.5.1.1 Division of Nuclear Power (NUC PR)

The Division of Nuclear Power is responsible for the safe, efficient, and environmentally sound operation and maintenance of all TVA nuclear electric generating plants and will have this responsibility for the CRBRP. The organizational structure is shown in Figure 13.1-1.

The Director of Nuclear Power administers the work of the division and is assisted by the Assistant Director; Manager, Nuclear Production; Manager, Technical Support; and supporting staffs.

The Assistant Director assists the director in managing the division and is responsible for directing the work of division managers in the development and maintenance of programs to implement division policy.

The Manager, Nuclear Production is responsible for nuclear production activities including preliminary operations, preoperational testing, power generation, and field services at TVA nuclear generating plants. He will direct these activities of the CRBRP.

The Manager, Technical Support, is responsible for developing and maintaining written programs and providing direct technical support to the TVA nuclear plants in the areas of maintenance and engineering support, quality assurance and compliance, emergency preparedness and protection, industrial safety, and training. He supervises the Manager, Maintenance and Engineering, and the Chiefs of the Quality Assurance and Compliance Branch, the Emergency Preparedness and Protection Branch, and the Training Branch. He will assume these responsibilities for the CRBRP.

13.1.1.5.1.2 Other TVA Organizations

The TVA organization is shown in Figure 1.4-2. These offices/divisions will be available to provide support services to the CRBRP as required.

13.1.1.5.2 Project Technical Support

Project technical support for the operation of the CRBRP will be provided to the Division of Nuclear Power in TVA by the CRBRP Project Office, W-LRM, and Burns and Roe. Areas of support will be in accordance with the responsibilities described in Section 1.4.

13.1.2 Operating Organization

13.1.2.1 Plant Organization

The plant organizational chart is shown in Figure 13.1-2. The principal groups that function directly under the supervision of the Plant Manager and Assistant Plant Managers are the Plant Operations Section, the Plant Engineering Section, the Plant Maintenance Sections (Mechanical, Electrical and Instrument), and the Health and Safety Services. The CRBRP organization follows the pattern developed through experience and used at TVA nuclear generating plants.

Plant employees will be selected primarily from existing TVA plant staffs and NUC PR's central office. Personnel qualifications shall be in accordance with NRC Regulatory Guide 1.8-1-R-1977.

13.1.2.1.1 <u>Operations Section</u>

The Operations Section will be responsible for all plant operations. It will provide operating personnel to support the preoperational testing, fuel loading, startup testing, startup, and plant operation. It will be responsible for coordinating and scheduling the training program for all operations personnel. It will provide the nucleus of emergency teams such as the plant rescue and fire-fighting organizations.

The Operations Section will be under the direction of the Operations Supervisor who will hold a valid NRC Senior Reactor Operator (SRO) license. He will be assisted by two inline Assistant Operations Supervisors; each will hold a valid NRC SRO license.

There will be five shift crews within the Operations Section to provide 24-hour coverage for operating the plant. The minimum shift crew when the reactor is operating will consist of the Shift Engineer who will hold an NRC SRO license, one Assistant Shift Engineer who will hold an NRC SRO license, two Unit Operators; each will hold an NRC Reactor Operator (RO) license, one Shift Technical Advisor (STA), and two Assistant Unit Operators. One Health Physics Technician and one Radiochemical Laboratory Analyst will also be assigned to each shift. Additional operators will be assigned as necessary. Plant management and technical support will be present or on call at all times.

13.1.2.1.2 <u>Engineering Section</u>

The Engineering Section will be under the direction of the Engineering Section Supervisor. He will be assisted by a complement of engineers. The Engineering Section will be responsible for providing technical direction and staff assistance in the areas of nuclear, mechanical, and chemical engineering. Responsibilities of this section will include plant and equipment performance tests, inplant fuel management, waste management and chemistry control.

The Engineering Section will carry out a comprehensive program of tests, studies, and investigations for the purpose of monitoring the reactor, engineered safeguards, and plant operating conditions to assure compliance with the operating license and to improve the efficiency of the plant. This will include the coordination of the surveillance test program with other plant sections.

13.1.2.1.3 Mechnical Maintenance Section

The Mechanical Maintenance Section will be under the direction of the Mechanical Maintenance Section Supervisor. He will be assisted by an inline Assistant Maintenance Supervisor.

The Mechanical Maintenance Section will be responsible for mechanical maintenance work and inspections in the plant. This will include scheduling and conducting periodic inspections and tests on the systems assigned to this section associated with the reactor and engineered safeguards, as required by the operating license. This section will develop and carry out a preventive maintenance program that assures that the repair and replacement of parts is consistent with the intent of applicable codes and basic requirements of the original equipment. A record file will be maintained by the section on all equipment, inservice tests, inspections, and maintenance reports.

13.1.2.1.4 Electrical Maintenance Section

The Electrical Maintenance Section will be under the direction of the Electrical Maintenance Section Supervisor. He will be assisted by an inline Assistant Maintenance Supervisor.

The Electrical Maintenance Section will be responsible for electrical maintenance work and inspections in the plant. This will include scheduling and conducting periodic inspections and tests on the systems assigned to this section associated with the reactor and engineered safeguards, as required by the technical specifications and operating license. This section will develop and carry out a preventive maintenance program that assures that the repair and replacement of parts is consistent with the intent of applicable codes and basic requirements of the original equipment. A record file will be maintained by the section on all equipment, inservice tests, inspections, and maintenance reports.

13.1.2.1.5 Instrument Maintenance Section

The Instrument Maintenance Section will be under the direction of the Instrument Maintenance Section Supervisor. He will be assisted by two inline Assistant Maintenance Supervisors.

The Instrument Maintenance Section will be responsible for instrument maintenance work and inspections in the plant. This will include scheduling and conducting periodic inspections and tests on the systems assigned to this section associated with the reactor and engineered safeguards, as required by the operating license. This section will develop and carry out a preventive maintenance program that assures that the repair and replacement of parts is consistent with the intent of applicable codes and basic requirements of the original equipment. A record file will be maintained by the section on all equipment, inservice tests, inspections, and maintenance reports.

13.1.2.1.6 Health and Safety Services

This department will consist of two groups which provide services in the Health and Safety Areas. Those units which will report administratively offsite and functionally onsite to the Assistant Plant Manager for Health and Safety Services will consist of Medical, Health Physics, Security, and Power Stores. They will be responsible for Medical Services, Radiological Protection, Plant Industrial Protection, and the procuring, storing, and issuing of items/components needed for the operation and maintenance of the plant. Those units which will report both administratively and functionally onsite to the same Assistant Plant Manager will consist of Industrial Safety and Fire Protection, Compliance, Quality Assurance, Document Control, and Administrative Services. They will have the responsibility for Plant Safety, and the many facets relating to Quality Control and Administrative Control, and ensuring compliance with industry and federal standards and requirements.

13.1.2.2 Personnel Functions, Responsibilities, and Authorities

During normal plant operations, the plant manager will be responsible for all plant activities. In the event of absences, incapacitation of personnel, or other emergencies, the following persons will be responsible in the order listed for all plant activities:

- Plant Manager
- Assistant Plant Managers
  - Engineering and Operations
  - Maintenance
- Plant Operating Supervisor

13.1.2.2.1 Plant Manager

The Plant Manager will have direct responsibility for all plant activities. He will be responsible for safeguarding the general public and plant employees from hazards associated with the operation of the CRBRP through implementation of the TVA hazard control standards and requirements, applicable DOE and NRC rules and regulations, and plant procedures; and for adherence to all requirements of the operating license. He will receive direction and supervision from the Director, Division of Nuclear Power and staff assistance from the Division of Nuclear Power Central Office.

13.1.2.2.2 Assistant Plant Manager

There are three Assistant Plant Managers: Assistant Plant Manager for Engineering and Operations; Assistant Plant Manager for Maintenance; and Assistant Plant Manager for Health and Safety Services. These managers will assist the Plant Manager in planning, coordinating, and directing the plant activities under their control. In the absence of the Plant Manager an Assistant Plant Manager will be responsible for management of plant activities as indicated in Section 13.1.2.2.

13.1.2.2.3 Operations Supervisor

The Operations Supervisor will be responsible for the safe and efficient operation of the plant in accordance with the operating license, approved procedures and TVA hazard control standards and requirements. He will be responsible for the preparation and maintenance of up-to-date operating instructions and the preparation of operating records. He will also be responsible for operator training programs and operating personnel schedules and will be charged with the responsibility of keeping the Plant Manager fully informed in all matters of operating significance.

13.1.2.2.4 Engineering Supervisor

The Engineering Supervisor will serve as supervisor of the Engineering Section and as a staff engineer in providing engineering advice and assistance to the Plant Manager. He will be responsible for initiating, planning, and coordinating the technical training programs. His experience and training must provide him with a good understanding of nuclear reactor technology, hazards, safeguards, licensing requirements, and a knowledge of the control systems used in a nuclear plant. He will be responsible for analysis of the performance of the reactor and turbine cycle and associated equipment during the test, startup, and operation of the plant.

13.1.2.2.5 Mechanical Maintenance Supervisor

The Mechanical Maintenance Supervisor will be responsible for all mechanical maintenance work and inspections in the plant. He will be responsible for maintaining safe working conditions for his employees and for their adherence to safe working practices. He will be assisted in his work by an Assistant Supervisor with experience in mechanical maintenance. He will also be assisted by foremen of the various crafts within the section and engineers who will be assigned to the plant as the workload demands. The Mechanical Maintenance Supervisor shall have a thorough knowledge of the operation and maintenance of all plant mechanical equipment.

13.1.2.2.6 Electrical Maintenance Supervisor

The Electrical Maintenance Supervisor will be responsible for all electrical maintenance work and inspections in the plant. He will be responsible for maintaining safe working conditions for his employees and for their adherence to safe working practices. He will be assisted in his work by an Assistant Supervisor with experience in electrical maintenance. He will also be assisted by foremen of the electrical craft within the section and engineers who will be assigned to the plant as the workload demands. The Electrical Maintenance Supervisor shall have a thorough knowledge of the operation and maintenance of all plant electrical equipment.

13.1.2.2.7 Instrument Maintenance Supervisor

The Instrument Maintenance Supervisor will be responsible for all instrument maintenance work and inspections in the plant. He will be responsible for maintaining safe working conditions for his employees and for their adherence to safe working practices. He will be assisted in his work by two assistant supervisors with experience in instrument maintenance. He will also be assisted by foremen of the Instrument Mechanics within the section and engineers who will be assigned to the plant as the workload demands. The Instrument Maintenance Supervisor shall have a thorough knowledge of the operation and maintenance of all plant instrumentation.

13.1.2.3 Shift Crew Composition

Normal Operations

The Shift Engineer on duty, a SRO, will be in direct charge of the plant including startup, operation, and shutdown of the reactor and ancillary systems. He may institute immediate action in any given situation to eliminate difficulties or remove equipment from service to preclude violation of the operating license or to avert possible injury or undue radiation exposure of personnel.

The Assistant Shift Engineer, a SRO, will be under the immediate supervision of the Shift Engineer. He will follow established procedures in doing his work. However, if a particular situation is not covered by a procedure, he may seek advice from the Shift Engineer; or, if the situation is critical, he may use his own judgment to prevent damage to equipment, injury to personnel, or undue radiation exposure of personnel. The duties of the shift supervisor as identified by NRC will be performed by the assistant shift engineer. The normal work station for the shift supervisor (assistant shift engineer) will be in the control room, but he periodically makes inspections of the plant equipment. He will immediately go to the control room during emergency situations. He will remain in the control room at all times during the accident situations to direct the activities of the unit operator unless formally relieved of this function by the shift engineer.

The shift technical advisor will report to the shift supervisor in the control room during off-normal reactor plant conditions. The role of the shift technical advisor will be to serve in the advisory capacity to the shift supervisor and not to assume command or control functions. The shift supervisor may choose to direct the shift technical advisor to perform his advisory role from either the control room or the onsite technical support center, or the shift supervisor may direct the shift technical advisor to serve as a liaison between technical support personnel manning the onsite technical support center and the shift supervisor.

Routine duties and assignments of the shift technical advisor will include matters involving engineering evaluation of day-to-day plant operations from a safety point of view.

The Unit Operator who will hold a RO License will be under the immediate supervision of the Assistant Shift Engineer and the general supervision of the Shift Engineer. He will follow established procedures in performing his work. In emergencies when there is not time to get advise from his supervisor he may deviate from established procedures to prevent damage to equipment, injury to personnel, or undue radiation exposure of personnel. He will be responsible for the safe and efficient operation of the plant from the control room or from local control stations.

The Assistant Unit Operator will be under the immediate supervision of the Unit Operator and the general supervision of the Assistant Shift Engineer. He will follow established operating instructions in doing his work and will not deviate from those instructions except as directed. He will perform assigned routine inspections and manipulative operations without close supervision. He will assist in the operation and perform work requirements within defined areas such as the Control Building, Reactor Containment Building, Reactor Service Building, Turbine Generator Building, Diesel Generator Building, Intermediate Building, Steam Generator Building, and Intake Structure.

When on shift, the Radiochemical Laboratory Analyst will be under the functional supervision of the Shift Engineer. These duties consist of periodic sampling of the various systems, such as feedwater and main steam, water makeup, waste condensate, and periodic monitoring of the primary and secondary sodium coolant.

When on shift, the Health Physics Technician will be under the functional supervision of the Shift Engineer. He will perform routine radiation surveys, personnel monitoring activities, and other assigned duties. He will keep the Shift Engineer informed of radiation hazards and performs special surveys as requested.

Minimum Shift Crew Composition

A minimum shift crew will be established for specific plant conditions. These requirements are given in Section 16.6.

13.1.3 Qualification Requirements for Nuclear Plant Personnel

All personnel at the CRBRP will be required to obtain and maintain qualification standards in accordance with NRC Regulatory Guide 1.8-1-R-1977. The personnel selection and training program that assures fulfillment of these qualification requirements also satisfies NRC Regulatory Guide 1.8-1R-1977. In addition, the operation of the CRBRP will be conducted in accordance with NRC Regulatory Guide 1.33-R2-1978, TMI Action Plan Items, I.C.3 of NUREG-0694-1980, and I.A.1.1 & I.A.1.3 of NUREG-0737-1980.

13.1.3.1 Qualifications of Plant Personnel

The positions of the plant staff have not yet been filled.

FIGURE 13.1-1

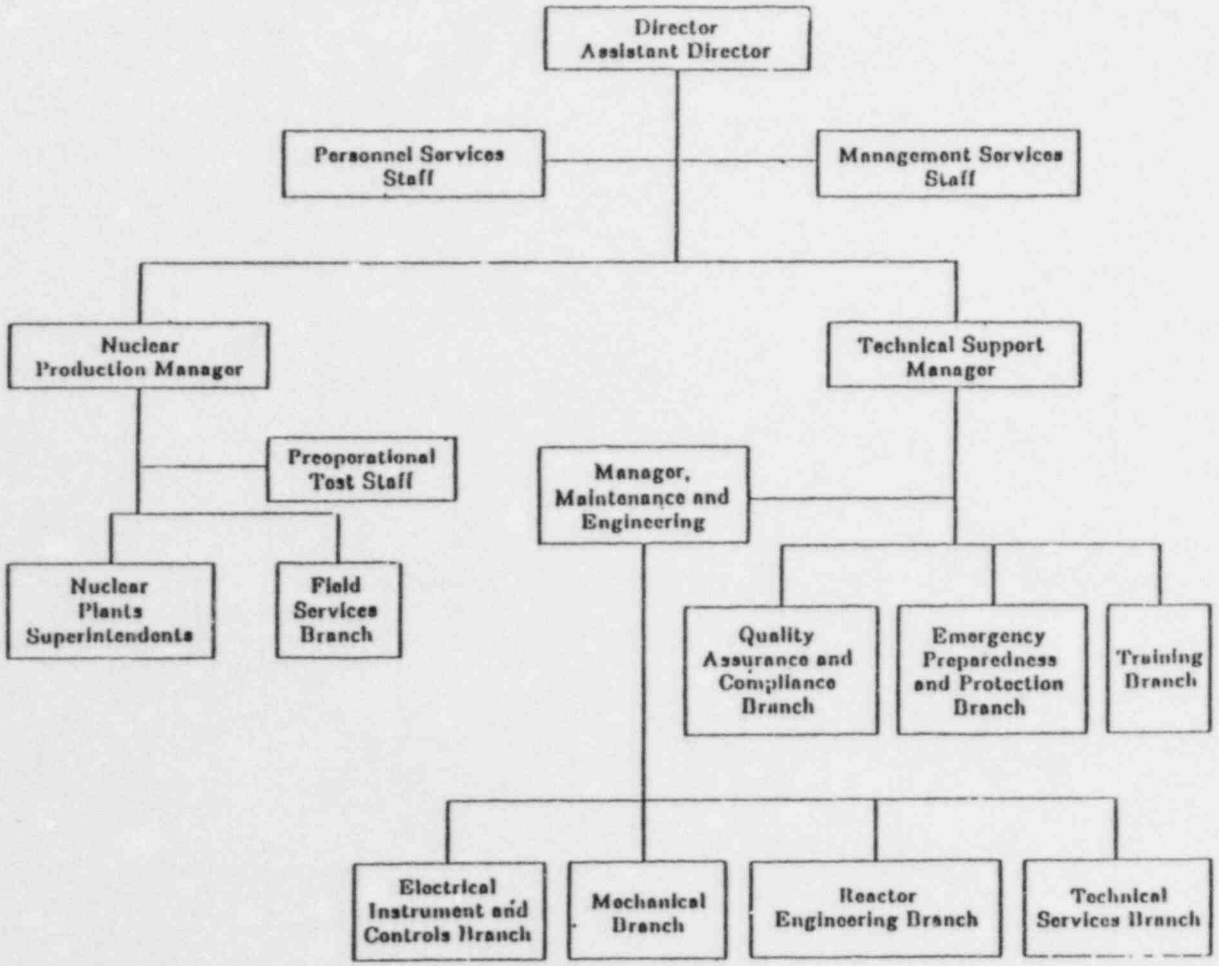

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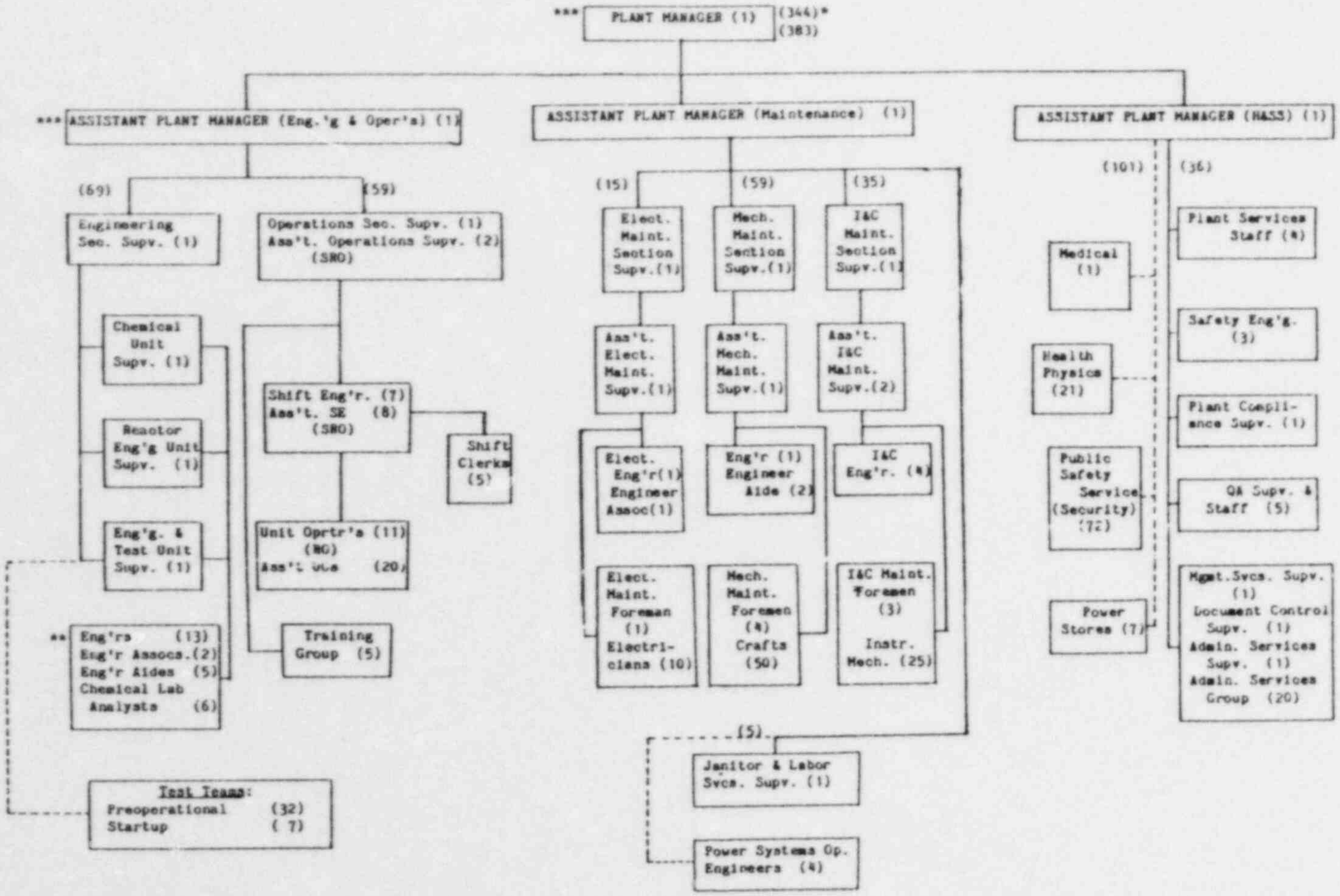
FIGURE 13.1-2
CRBRP ORGANIZATIONAL CHART



NOTE: Dash line indicates Functional Supervision;
— Solid line indicates Functional & Administrative.
— Information based on latest staffing plans.
— The numbers indicate requirements for normal operation.
— There will be five operating shifts.

* Test Teams (39) will transfer offsite 1-2 years after criticality.
** The Shift Technical Advisors (STA) are included here.
*** Will take training but not necessarily the NRC Exam.

2/2/82

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13.2 TRAINING PROGRAM

13.2.1 Program Description

An onsite formal training program will be conducted prior to initial fuel loading.

The basic objectives of the training program are:

a. To assure that all plant personnel are properly trained and qualified to perform their assigned tasks in a safe and efficient manner.

b. To assure that the CRBRP is operated in accordance with NRC regulatory requirements and guidelines.

c. To assure that all training is formally documented.

d. To meet or exceed NRC licensing requirements.

In achieving these objectives, individual training needs will be established by comparing job requirements with individual experience.

The training program, as it is initially planned, will be approved by the Manager, Technical Support, after being approved by the Plant Manager. The properly planned and approved program will ensure that the content and the intent of the training program provides the necessary training for personnel associated with reactor operations. The program will be designed to train personnel both with and without previous nuclear experience. It will consist of both formal classroom-type lectures, self-study assignments, on-the-job training, and progress evaluation. This program will be the mechanism to ensure that all members of the plant operating staff are qualified for the assigned position prior to initial fuel loading.

The effectiveness of the training program will be evaluated by the performance of employees on TVA and NRC examinations in carrying out their assigned duties and by periodic evaluations of performance by supervisors. In addition, periodic audits of the training program will be performed by designees within the Office of Power, but outside the Division of Nuclear Power.

13.2.1.1 Program Content

At the time of manning the CRBRP, TVA should have highly trained nuclear plant operating personnel at the Browns Ferry, Sequoyah, Watts Bar, and Bellefonte Nuclear Plants. These plants will be the primary source of personnel for the CRBRP.

Those positions at the CRBRP which require an NRC licensed SRO will be filled by personnel who have or are eligible to sit for an NRC SRO license on a light-water reactor power plant. Other positions will be filled by personnel within the TVA organization as available and selected from competent applicants from outside TVA. All CRBRP personnel will be given comprehensive training to produce personnel who have that combination of education, experience, and skills commensurate with their level of responsibility. This will provide reasonable assurance that decisions and actions during all normal and off-normal conditions will be such that the plant is operated in a safe and efficient manner.

The program, as being developed, complies with 10 CFR 55-1980, TMI Action Plan Items, I.A.1.1, I.A.2.1, I.A.2.3, I.A.3.1, and II.B.4 of NUREG-0737-1980, NRC Regulatory Guides 1.149-1981 and 1.8-1-R-1977, Item I.A.4.2 of NUREG-0718-R2-1982, and BTP CMEB 9.5-1-R3-1981.

Emphasis is placed on simulator use as an integral part of the training program. A plant-specific simulator will be available for the CRBRP to be located onsite and operational at the time of commencing preoperational testing. It will serve as a device for procedure checkout as well as a vital component of the operator training program. The training program for all candidates seeking NRC SRO and RO licenses will include significant time at the simulator. Selected personnel in the non-NRC licensed category will also receive training at the simulator. The simulator will accurately reproduce the general operating characteristics of the CRBRP, and the arrangement of the instrumentation and controls of the simulator will closely parallel that of the CRBRP.

The program will provide training for the following categories:

a. Individuals with no previous power plant experience
b. Individuals with no previous nuclear power plant experience
c. Individuals who have had nuclear experience but not NRC licensed
d. Individuals who hold, or have held, an NRC license for a LWR facility or a facility comparable to CRBRP
e. Individuals who will obtain an NRC SRO or RO 'cold' license
f. Individuals who will obtain an NRC SRO or RO 'hot' license
g. Individuals who will take the training for an NRC license but may not take the NRC exam [e.g. The Plant Manager or Assistant Plant Manager (Engineering and Operation) and the Shift Technical Advisor]
h. Individuals who will not be taking the license-type training, i.e., all plant staff whose positions do not require an NRC license

Table 13.2-1 includes detailed information on the training program, such as: Subject matter of each course, the duration of the course, the organization teaching the course, and the position titles for which the course will be given. Figure 13.2-1 presents a proposed training schedule for the CRBRP staff which is in accordance to NRC Regulatory Guide 1.8-1-R-1977. It is planned that the following personnel will obtain a 'cold' license in accordance with the requirements of 10 CFR 55-1980 before initial fuel loading: SRO license for the Operations Supervisor, at least five Shift Engineers, and at least five Assistant Shift Engineers; and obtain RO licenses for at least ten Unit Operators. The Plant Manager or the Assistant Plant Manager will obtain the training required for an SRO license. Since the 'cold' license will be obtained prior to criticality, that part of the training consisting of actual startups will be performed at a research or power reactor and the control manipulations will be performed at the CRBRP- Specific simulator substituting for the real plant.

Under NRC procedures for determining the eligibility of an applicant to take a cold examination, the applicant is considered to have had extensive operating experience at a comparable facility if one of the following four conditions have been met:

a. Holding or having held an operator's or senior operator's license at a comparable licensed reactor facility. To date, NRC has considered any light water power reactor as comparable to any other light water power reactor. However, it is highly desirable that previously licensed individuals participate in a short course utilizing a nuclear power plant simulator similar to the facility for which the applicant will be seeking a license.

b. Determination of such experience as indicated in a. above at a comparable reactor facility not subject to NRC licensing (e.g., reactor facilities operated by the military services or owned by the Department of Energy (DOE).

c. The applicant has passed an NRC-administered written examination and operating test at a comparable licensed reactor facility after completing an NRC approved training program. Normally, the Commission issues a certification stating that the individual has met the requirements of an operator as set forth in 10 CFR 55-1980 in lieu of a license.

d. Certification of satisfactory completion of an NRC-approved training program which utilizes a nuclear power plant simulator as part of the program.

The non-NRC licensed personnel will also undergo training both prior to and after assignment to the CRBRP. The training program in which these employees participate prior to CRBRP assignment will include formal programs at TVA's Power Operations Training Center (POTC), TVA Administrative and Standards of Apprenticeship, or approved equals. After assignment to CRBRP the training will be commensurate with the individual's experience and job assignment and will follow the guidelines of Item I.A.2.2 of NUREG-0660-1980, and be conducted in accordance to NRC Regulatory Guide 1.8-1-R-1977.

The training of personnel for the position of Shift Technical Advisor (STA), a non-NRC licensed position, is specifically addressed because the STA serves the important function of technical advisor to the Shift Supervisor. The details of the STA training in the CRBRP Training Program when developed at the FSAR stage will be in accordance with TMI Action Item I.A.1.1 of NUREG-0737-1980.

A fire protection training program will be developed prior to fuel loading and will meet the guidelines of CMEB 9.5-1-R3-1981 as applicable to CRBRP, and include details such as interface requirements between the Plant

Operator and the Plant Constructor. The fire protection program of the Constructor will be the basis for the construction force fire-protection training. On occasions of interface between construction and operations in areas under control of operations, the involved construction personnel will be oriented on the applicable portion of the Operator's fire protection program. The fire prevention and protection program for plant operation will use the fire brigade system with properly trained personnel on each shift. The program will be formulated by the Safety and Fire Protection Engineer who will have had, prior to his assignment to the CRBRP Project, formal training in the areas of safety and fire protection engineering from an accredited school, college, or university. The post-CRBRP training will include plant-system familiarization and the techniques to be employed in preventing, detecting, control, and fighting of sodium and sodium-potassium fires.

The individual tasks and the various components of each position will be examined to identify the knowledge, skill, physical ability, duties, and responsibilities required for the individual to accomplish his or her job. In addition, the performance standards to be met in completing the task will be identified. This will be achieved by developing job descriptions for the positions and evaluating performance records of incumbents in similar positions before the positions at the CRBRP are filled.

The actual training program and schedule for the plant operating staff will be designed around the findings of a job/task analysis. Hence, the schedule is tentative and subject to change before submission as a part of the Final Safety Analysis Report.

13.2.2 Retraining Program

This information will be included in the FSAR.

13.2.3 Replacement Training

This information will be included in the FSAR.

13.2.4 Records

13.2.4.1 TVA

Official records of employee qualifications, experience, training, and retraining of each member of the plant organization are maintained in the official TVA Personal History Record (PHR) by the Division of Personnel. The PHR provides in a standardiz· arrangement, the information officially recognized in recording and sup, rting employee status. The PHR is maintained in current and acc ·ate status and is controlled as to availability. The material admitted to this record is restricted to items for which authenticity has been confirmed through established procedures; for example, official TVA forms, signed statements from the employee, and management representatives. The official records, as identified above, of each TVA employee on the CRBRP Staff will be maintained in the PHR.

13.2.4.2 Plant

Records supporting requests for NRC SRO and NRC RO licenses will be maintained in the plant master file. These records will include training courses attended, retraining classes, number of reactor startups, and other information necessary to ensure that training requirements have been met. Some of these records will be duplicated in the PHR.

A training file for each member of the plant organization will be maintained in the plant master file. Information regarding participation in training and retraining activities and records of employee participation in training activities leading to promotion to a higher level of competence will be maintained in this training file.

TABLE 13.2-1
PROPOSED TRAINING PROGRAM

Subject Areas & Composition	Duration	Organization Teaching	Position Receiving Training
Candidates for NRC-License			
I. LMFBR & CRBRP-Specific Technology o Basic differences between LWRs and LMFBRs o Principles of Operation o Design Features o General Operating Characteristics o Reactor Instrumentation & Control o Specific subjects in 10CFR55-1980, Sections 21 and 22, and NUREG-0737-1980 o Heat Transfer, Fluid Flow, & Thermodynamics o Use of plant systems to control or mitigate an accident in which the core is damaged o Reactor & Plant transients	6 weeks	Onsite by Staff specialists	Plant Manager Assistant Plant Managers Operations Section Supervisors Shift Engineers Assistant Shift Engineers Unit Operators (Although the STA is a non-NRC licensed position, the STAs will receive this training.)
II. Sodium & Sodium Potassium Handling			
A. Offsite 1. Sodium Systems 2. Sodium System Components 3. Engineering Properties of Sodium 4. Corrosion, Mass Transfer, and Chemical Instrumentation 5. Materials Selection 6. Sodium Instrumentation and Electrical Systems 7. Operational Consideration 8. Maintenance 9. Sodium Safety	2-3 weeks	Offsite at Vendors Training Center	Selected members of Manager's Staff and Operating Section.
B. Onsite 1. Sodium Systems 2. Sodium System Components 3. Engineering Properties of Sodium 4. Corrosion, Mass Transfer, and Chemical Instrumentation 5. Materials Selection 6. Sodium Instrumentation and Electrical Systems 7. Operational Consideration 8. Maintenance 9. Sodium Safety	4 weeks	Onsite by Selected Staff Members who attended the offsite course	Plant Managers Assistant Plant Managers Operations Section Supervisors Shift Engineers Assistant Shift Engineers Unit Operators (Those of above who did not take the offsite course.)
III. CRBRP Plant Systems Lecture Series Includes lectures on details of each of the plant systems	20 weeks	Onsite by Engineers from Engineering Section who have become specialists in the systems	All personnel for plant positions who receive training for NRC license

TABLE 13.2-1 (Cont'd.)
PROPOSED TRAINING PROGRAM

Subject Areas & Composition	Duration	Organization Teaching	Position Receiving Training
Candidates for NRC License (Cont'd.)			
IV. Observation Training 1. LMFBR Plant Familiarization 2. Operating Evaluations 3. Liquid Metal Systems 4. Inert Gas Systems 5. Characteristics of an Operating LMFBR 6. FFTF Simulator (for LMFBR Familiarization)	up to 6 months	Offsite at an operating LMFBR	All personnel for plant positions who will receive training for NRC license.
V. Health Physics 1. Radiation Exposure Limits 2. Radiation 3. Special Work Permits 4. Contamination Control 5. Airborne Activity 6. Injuries in Contamination or Radiation Areas 7. Handling, Storage, and Transfer of Radioactive Materials 8. Safe Practices 9. REP for CRBRP	2 weeks	Onsite by Health Physics Supervisor	Plant Managers Assistant Plant Managers Operations Section Supervisors Shift Engineers Assistant Shift Engineers Unit Operators
VI. Work/Study Assignments TVA Management/Superv. Skills TVA Administrative Controls	1 week	Offsite personnel from TVA's Central Office	Selected positions such as Plant Managers, Assistant Plant Managers, Operations Section Supervisors, Shift Engineers, Assistant Shift Engineers, Unit Operators --
Operating Practices o Reactor Startups, Operating Instruction Preparation Sodium Load, Preoperational Tests Fuel Load, Plant Procedures	18 months	Onsite by Training Coordinator	Plant Managers, Assistant Plant Managers, Operations Section Supervisors, Shift Engineers, Assistant Shift Engineers, Unit Operators --
o CRBRP-Specific Simulator Training	200 hours		
o CRBRP-Control Room Operational	480 hours		
o 3 months on shift as licensed Operator	3 months		(For Candidates for 'hot' license only)
o Self-directed study	6 months		
VII. Pre-NRC License Examination		Onsite by Training Coordinator	All who are taking NRC examination
o Operator Administered Exam	1 week		
o Inplant Briefing	1 week		
o Pre-license examination review	2 weeks		

TABLE 13.2-1 (Cont'd.)

PROPOSED TRAINING PROGRAM

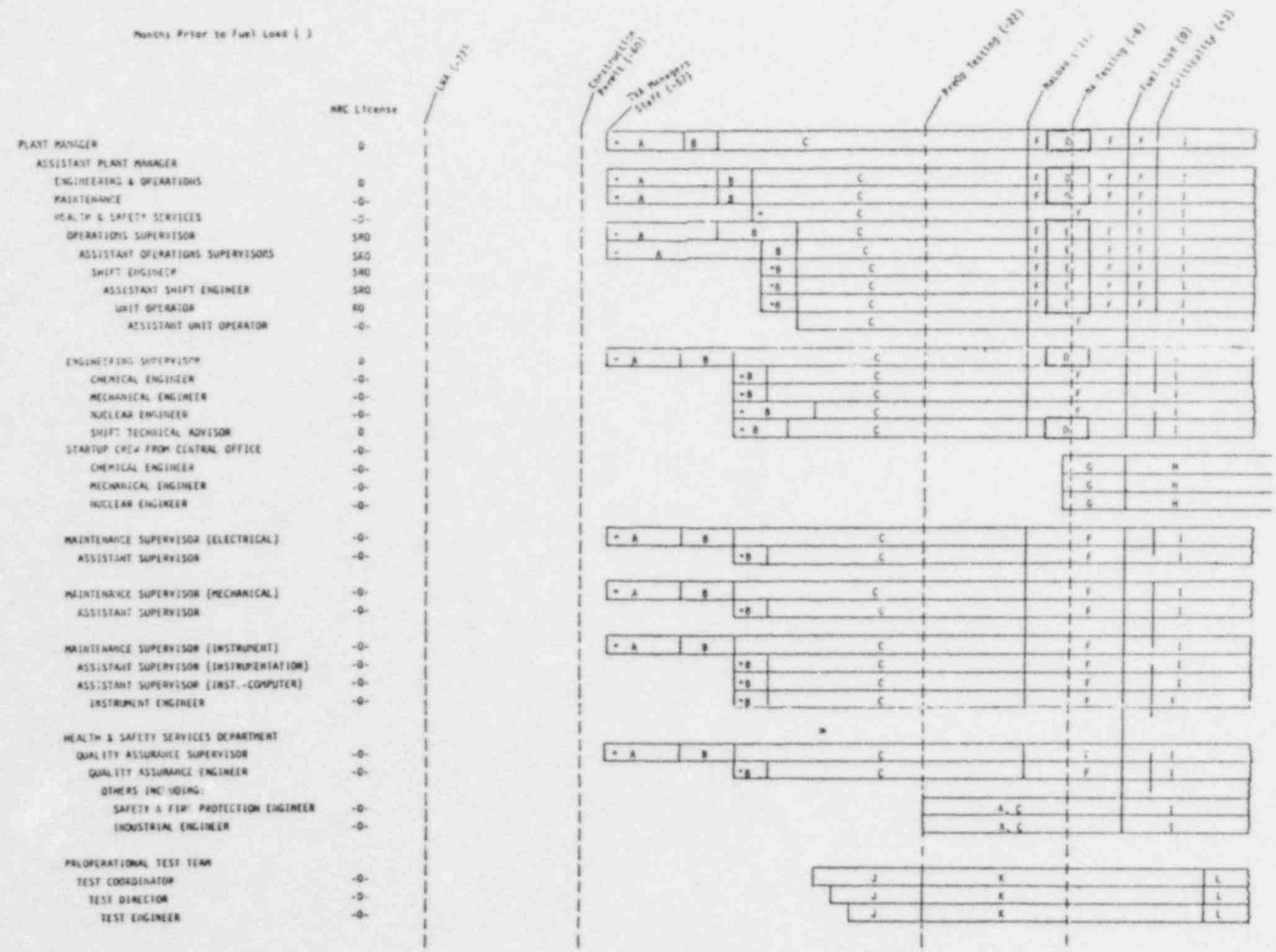
Subject Areas & Composition	Duration	Organization Teaching	Position Receiving Training
Candidates for NRC-License (Cont'd.)			
VIII. NRC License Examination Written Operating (to include simu-lator examination)	1 week	Onsite by Training Coordinator	Only those on the Manager's Staff and in Operating Section that are obtaining NRC license
IX. General Employee Training 1. Familiarization with Plant-Specific administrative procedures and plans to enable trainee to demonstrate know-ledge in those areas of trainee's specialty. 2. Radiological Health & Safety 3. Industrial Safety 4. Plant-Controlled Access Areas and Security Procedures 5. Use of Protective Clothing and Equipment 6. Plant Quaity Assurance Program 7. Fire Protection Program 8. Plant Cleanliness and Housekeeping Requirements 9. General Description of Plant and Facilities 10. CRBRP REP	4 weeks	Onsite personnel by specialists in subject disciplines.	Plant Managers, Assistant Plant Mana-gers, Operations Section Supervisors, Shift Engineers, Assistant Shift Engi-neers, Unit Operators
X. CRBRP Emergency Instructions	1 week	Onsite-by staff specialist	Plant Managers, Assistant Plant Mana-gers, Operations Section Supervisors, Shift Engineers, Assistant Shift Engi-neers, Unit Operators
XI. Fire Brigade Training	1 week	Onsite-by staff specialist	Plant Managers, Assistant Plant Mana-gers, Operations Section Supervisors, Shift Engineers, Assistant Shift Engi-neers, Unit Operators
Non-NRC Licensed Personnel			
I. LMFBR & CRBRP-Specific Technology Contents of course for non-licensed personnel will be tailored to specific disciplines	3 weeks	Onsite by specialists	Selected technical and supervisory personnel
II. Sodium & Sodium-Potassium Handling A. Offsite 1. Sodium Systems 2. Sodium System Components 3. Engineering Properties of Sodium 4. Corrosion, Mass Transfer, and Chemical Instrumentation 5. Materials Selection 6. Sodium Instrumentation and Electrical Systems 7. Operational Consideration 8. Maintenance 9. Sodium Safety	3 weeks	Offsite at Vendor's Training Center	Selected technical and supervisory personnel

TABLE 13.2-1 (Cont'd.)
PROPOSED TRAINING PROGRAM

Subject Areas & Composition	Duration	Organization Teaching	Position Receiving Training
Non-NRC Licensed Personnel (Cont'd.)			
B. Onsite 1. Sodium Systems 2. Sodium System Components 3. Engineering Properties of Sodium 4. Corrosion, Mass Transfer, and Chemical Instrumentation 5. Materials Selection 6. Sodium Instrumentation and Electrical Systems 7. Operational Consideration 8. Maintenance 9. Sodium Safety	4 weeks	Onsite by Engineers from Engineering Section who attended the offsite course	Selected Technical and Supervisory personnel
III. CRBRP Plant Systems Lecture Series Includes lectures on plant systems details	up to 20 weeks	Onsite by Engineers from Engineering Section who have become specialists in the systems	Selected personnel depending on their assigned position
IV. Health Physics & Safety	2 weeks	Onsite by the Health Physics Supervisor	All plant personnel including those on temporary assignment
V. Observation Training Training at an Operating LMFBR with content oriented to specific disciplines	up to 6 months	Offsite at an operating LMFBR	Selected members of maintenance supervision and of technical disciplines from the Engineering Section
VI. Onsite Work/Study Assignments o CRBRP Familiarization & Orientation o Developing Plant Procedures o Studying Plant Layout & Design o Technical Training in areas of Chemistry, Nuclear, Electrical, Instrumentation and Control o Basic Simulator Training o Supervisory Training TVA Management/Supervisory Skills TVA Administrative Controls o CRBRP-Specific Simulator Exercises o Participation in Preoperational & Startup Testing	52 weeks	Onsite by staff specialists	Selected technical and supervisory personnel
VII. General Employee Training 1. Familiarization with Plant-Specific administrative procedures and plans to enable trainee to demonstrate knowledge in those areas of trainee's specialty. 2. Radiological Health & Safety 3. Industrial Safety	4 weeks	Onsite personnel by specialists in subject disciplines	All plant personnel including those on temporary assignment

TABLE 13.2-1 (Cont'd.)
PROPOSED TRAINING PROGRAM

Subject Areas & Composition	Duration	Organization Teaching	Position Receiving Training
Non-NRC Licensed Personnel (Cont'd.)			
4. Plant-Controlled Access Areas and Security Procedures 5. Use of Protective Clothing and Equipment 6. Plant Quaity Assurance Program 7. Fire Protection Program 8. Plant Cleanliness and Housekeeping Requirements 9. General Description of Plant and Facilities 10. CRBRP REP			
VIII. CRBRP Emergency Instructions	1 week	Onsite-by staff specialist	All plant personnel including those on temporary assignment
IX. Fire Brigade Training	1 week	Onsite-by staff specialist	All permanently assigned plant personnel who may at sometime be a member of the fire brigade



LEGEND:

- * - Assignment to CRBRP Project
- A - Onsite Familiarization
- B - Assignment to Operating LMFBR
- C - Onsite Training Program (See Table 13.2-1 for details)
- D - Intensive Operator Licensing Activities - W/O NRC Exam
- E - Intensive Operator Licensing Activities - With NRC Exam
- F - Sodium Load & Fuel Loading
- G - Assignment Onsite for Startup Familiarization
- H - Assignment As Member of Startup Crew Assisting Operations with Assigned Duties
- I - Startup Tests and Operation During Demonstration Period
- J - Assigned Onsite for Familiarization and Preparation of Preoperational Test Procedures
- K - Performing Preoperational Tests
- L - Return to Central Office

FIGURE 13.2-1
PROPOSED TRAINING SCHEDULE

13.3 EMERGENCY PLANNING

13.3.1 General

The Clinch River Breeder Reactor Plant Radiological Emergency Plan (CRBRP-REP) will be developed to provide protective measures for Plant personnel, and to protect the health and safety of the public in the event of a radiological emergency resulting from an inplant accident or an accident involving transportation of radioactive waste from Clinch River Breeder Reactor Plant. This plan fulfills the requirements set forth in 10 CFR 50-1982 that an emergency plan be included in the Preliminary Safety Analysis Report and be developed in accordance with the Nuclear Regulatory Commission (NRC) and Federal Emergency Management Agency (FEMA) guidance. As specified in NUREG-0654-R1-1980, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, the CRBRP-REP will ensure that:

1. Adequate measures are taken to protect employees and the public.

2. All individuals having responsibilities during an accident are properly trained.

3. Procedures exist to provide the capability to cope with a spectrum of accidents ranging from those of little consequence to major core melt.

4. Equipment is available to detect, assess, and mitigate the consequences of such occurrences.

5. Emergency action levels and procedures are established to assist in making decisions.

The Radiological Emergency Plan will consist of the CRBRP-REP and appendices.

These documents are briefly described below. The actual CRBRP-REP will be submitted as a separate document prior to fuel loading.

13.3.1.1 CRBRP-REP

This document will address general organizational responsibilities, capabilities, actions, and guidelines for TVA and project personnel during a radiological emergency.

13.3.1.2 Appendices

Specific information on each of the TVA emergency centers (for the CRBRP these centers will serve as Emergency Operations Facilities) will be included as appendices to the CRBRP-REP. These appendices will detail facility features, capabilities, equipment protective actions, and responsibilities. The CRBRP-REP, together with the appendices, will describe the methods TVA will use to:

1. Detect an emergency condition

2. Evaluate the severity of the problems

3. Notify Federal, State, and local agencies of the condition

4. Activate emergency organizations

5. Evaluate the possible offsite consequences

6. Recommend protective actions for the public

7. Mitigate the consequences of the accident

Since TVA has no authority outside the plant boundaries, State and local radiological emergency plans will support the CRBRP-REP. State and local agencies are responsible for ordering and implementing actions offsite to protect the health and safety of the public. The plans for State and local agencies will be incorporated as Appendix F to the CRBRP-REP.

13.3.1.3 Additional TVA Plans and Procedures

The CRBRP-REP will compliment the following:

1. Spill Prevention Control and Countermeasure Plan

2. Physical Security Plans (which include fire protection)

3. Plant Equipment Normal, Abnormal, and Emergency Operating Instructions

4. Plant Surveillance Instructions

13.3.1.4 Implementing Procedures

Specific procedures will be developed to ensure that accidents are properly evaluated, rapid notifications made, and assessment and protective actions performed. These procedures will be compiled in the Clinch River Breeder Reactor Plant Implementing Procedures Document (CRBRP-IPD).

Plant instructions for normal and emergency system operation and control will exist but will not be included in the CRBRP-IPD. These plant operating instructions will be designed to ensure that operators implement the procedures specified in CRBRP-IPD.

13.3.1.5 State Radiological Emergency Plan

The State of Tennessee Radiological Emergency Plan as well as the plan for that portion of the State of North Carolina within the 50-mile ingestion pathway will be included in Appendix F of the CRBRP-REP.

13.3.2 Emergency Organization

The plant manager will be in charge of all activities at the site and will have assistant managers to share his responsibilities as he deems necessary. The minimum staffing requirements for operation will be found in the Plant

Technical Specifications. The responsibilities of the minimum staff under normal operations will be as outlined in the PSAR. Under emergency conditions the responsibilities of the minimum staff will be unchanged. The site emergency organization will augment the shift operations crew in accordance with NUREG-0654-R1-1980, Table B-1. If members of the site emergency organization are not present when an emergency occurs, the shift engineer on duty will be designated the Site Emergency Director until relieved by the plant manager or his alternate. The duties and responsibilities of the various plant supervisors concerning plant emergencies will be outlined in the CRBRP-REP.

An on-site Technical Support Center (TSC) will be provided in accordance with 10 CFR 50 Appendix E-1982 and NUREG-0696-1981, "Functional Criteria for Emergency Response Facilities". The TSC will perform the following functions:

- Monitor plant operations and provide technical advice and overall plant management from a non-control room location

- Supply sufficient plant information for analysis of plant status and extent of any plant damage

- Record sufficient information to allow for a thorough review of plant incidents and to aid in plant recovery

- Provide communications with off-site personnel and provide for other peripheral functions not directly related to reactor operations

- Relieve control room congestion

In order to provide these functions, the TSC will be located in close proximity to the control room. Access to the TSC during an emergency will be limited to those persons identified in the REP. The TSC will provide adequate working space for the identified support personnel and required support equipment. The TSC will be capable of being staffed and functional within approximately 30 minutes of an emergency. Radiological protection from airborne radioactivity and plant sources will be provided.

An on-site Operational Support Center (OSC) will provide a location where support personnel can assemble and where support can be coordinated and will provide communication with the control room and the TSC. The location of the OSC will be an existing area of the plant. The OSC will have access to equipment necessary to support the plant emergency response.

The on-site emergency organization will be supported by the staffs of four emergency centers: the Central Emergency Control Center (CECC) Staff and the Division of Nuclear Power Emergency Center (DNPEC) Staff in Chattanooga, Tennessee; the Muscle Shoals Emergency Control Center (MSECC) Staff in Muscle Shoals, Alabama; and the Knoxville Emergency Control Center (KECC) Staff in Knoxville, Tennessee. The CECC will function and provide all services as a 'near-site' Emergency Operations Facility. In addition, the offsite emergency organization will be supported by other TVA organizations as may be required.

The director of each center is responsible for directing his staff in carrying out their respective responsibilities. He is delegated the authority during emergencies to locate, direct, and dispatch the personnel and equipment necessary to carry out his staff's responsibilities.

The purpose of the CECC and associated CECC staff is to provide the facilities and manpower for evaluating, coordinating, and directing the overall activities involved in coping with a radiological emergency.

During an emergency, the CECC Director and his staff will review the response to the emergency by TVA and the appropriate State agencies to ensure that an effective and cooperative effort is being made. The CECC Director, after consultation with the Office of Health and Safety CECC representative, is responsible for providing TVA's recommended protective actions to the appropriate State officials.

The CECC staff will coordinate with all other TVA emergency centers to ensure an effective TVA effort in response to an accident situation. The CECC staff will also provide an accurate description of the emergency situation for TVA management and public information. In addition, the CECC will coordinate with offsite Federal agencies, such as the Nuclear Regulatory Commission (NRC) and Department of Energy (DOE), to ensure availability of additional outside resources to TVA.

The DNPEC staff provides support services during a radiological emergency to the affected plant. Support services may be provided by utilizing any necessary manpower and equipment under the direct control of the Division of Nuclear Power. If the division is unable to provide adequate services or support, requests will be made for additional support to other TVA divisions, local agencies, or government installations as may be required.

The Muscle Shoals Emergency Control Center (MSECC) supports the CECC by performing environmental radiological monitoring and dose assessments and by recommending protective actions for the public to the CECC. In performing these functions, the MSECC assists the Tennessee Department of Public Health in evaluating the population exposures resulting from radiological emergencies. The MSECC staff directs offsite environmental monitoring for the Tennessee Department of Public Health and continues monitoring activities until a State Field Coordination Center is established to coordinate the offsite environmental monitoring effort. The MSECC will continue to evaluate the need for monitoring assistance to the Tennessee Department of Public Health. The State may request assistance from the appropriate DOE Operations Office in accordance with Interagency Radiological Assistance Plan (IRAP) for additional support. The MSECC will monitor the radiation protection problems in the plant during emergencies to provide guidance, manpower, and equipment to the Plant Health Physicist as required to control and mitigate these problems.

The Knoxville Emergency Control Center (KECC) serves as the focal point for all essential support activities involving TVA Knoxville offices. The TVA Office of Engineering Design and Construction (OEDC), Division of Engineering Design (EN DES), has been delegated overall responsibility for the KECC and for providing technical support during and following a radiological emergency.

This section addresses the EN DES responsibilities for technical support during emergency conditions.

The KECC also serves as the communication center for other essential TVA offices such as the TVA Board of Directors, the General Manager, the Nuclear Safety Review Staff, and the Information Office.

The radiological emergency communications network will consist of a combination of commercial telephone circuits, radio, and microwave circuits. South Central Bell Telephone Company lines will be used as the primary means of communications during radiological emergency situations between plant, CECC, DNPEC, MSECC, KECC, and appropriate Federal and State agencies. This system will be augmented by the TVA Private Automatic Exchange (PAX).

The primary means of notification of plant and offsite personnel is the commercial telephone circuits. Additionally, pocket pagers are provided to certain key individuals in the emergency organization.

Figure 13.3-1 illustrates the relationship between the TVA emergency centers and depicts the interface among TVA, Federal, State, and local agencies.

13.3.3 Coordination With Offsite Groups

TVA will have agreements with other Federal agencies to assist in the evaluation and control of any radiological emergency. These agreements will include such agencies as the Department of Energy (DOE), Oak Ridge Operations Office, and the National Aeronautics and Space Administration (NASA), Marshall Space Flight Center. The CECC staff may request assistance from these outside agencies as required. The Site Emergency Director will be responsible for notification of NRC's regional office of Inspection and Enforcement.

Agreements will be made with the State of Tennessee, Tennessee Emergency Management Agency, to provide planning for emergencies at TVA nuclear facilities. This planning includes evacuation arrangements, traffic control, and support from other state agencies as required. The Clinch River Breeder Reactor Plant Radiological Emergency Plan will utilize the liaisons already established in developing the Browns Ferry, Sequoyah, Watts Bar, and Bellefonte Radiological Plans with the States of Alabama and Tennessee. The Tennessee Emergency Management Agency will notify the State of North Carolina and surrounding states and coordinate assistance from the various state agencies.

TVA will maintain liaison with the Tennessee Emergency Management Agency, particularly with respect to the availability of emergency services. The Tennessee Emergency Management Agency will inform these agencies of actions to be taken under their respective statutory authority and assist them in developing emergency procedures. TVA will provide any necessary training for local fire and police departments, ambulance services, and hospitals in radiological hygiene practices and recognition of raidological hazards. The attached Table 13.3-1 lists the organizations that will be participating in the Clinch River Breeder Reactor Plant Radiological Emergency Plan.

Arrangements will be made for an ambulance service to provide emergency service as required to the plant. Agreements will also be culminated between TVA and a nearby hospital to provide emergency treatment to irradiated or contaminated patients as required. TVA will assist in training ambulance attendants and hospital personnel in this type of treatment and will ensure that adequate equipment is made available as necessary. An agreement will also be made with the Radiation Emergency Assistance Center and Training Site (REAC/TS) Facility operated by the Oak Ridge Associated Universities (ORAU) for emergency treatment of severely contaminated or irradiated personnel.

13.3.4 Emergency Action Levels

TVA will utilize the following emergency classification:

1. Notification of Unusual Event

2. Alert

3. Site Emergency

4. General Emergency

This system of classification will be consistent with the system used by State and local emergency organizations.

A Notification of Unusual Event will provide early and prompt notification of minor events which could develop into or be indicative of more serious conditions which are not yet fully realized. The purposes of Notification of Unusual Event are to (1) assure that the first steps in activating emergency organizations have been carried out, and (2) provide current information on unusual events.

An Alert class will be indicated when events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. The purposes of the Alert class are to (1) assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and (2) provide offsite authorities current status information.

A Site Emergency will be declared when events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. The purposes of the Site Emergency class are to (1) assure that response centers are staffed, (2) assure that monitoring teams are dispatched, (3) assure that personnel required for evacuation of nearsite areas are at duty stations if the situation becomes more serious, and (4) provide current information for and consultation with offsite authorities and the public.

A General Emergency will be declared when events are in progress or have occurred which involve actual or imminent substantial core failure with potential for loss of containment integrity. The purposes of the General Emergency class are to (1) initiate predetermined protective actions for the public, (2) provide continuous assessment of information from the site and offsite, and (3) initiate additional measures as indicated by releases or potential releases of radioactivity.

Recognition of the emergency class which may result from many initiating conditions will be primarily a judgement matter for plant operating personnel. The initiating conditions used for recognizing and declaring the emergency class will be based on specific measured values or observable conditions defined as Emergency Action Levels (EAL). These can be combinations of specific instrument readings (including their rates of change), annunciator warnings, time periods certain conditions exist, etc. The specific instrument readings and parameters required for determination of these EAL's will be detailed in plant operating instructions. These EAL's will be used as thresholds for determining the emergency classifications.

13.3.5 Protective Measures

13.3.5.1 Plant

In the event of an unplanned release of radioactivity or sudden increase in radiation levels, it will be the responsibility of the Site Emergency Director to make the decision concerning the necessity for building and/or area evacuation. In arriving at this decision, the primary considerations will be personnel safety. The emergency siren will be used to initiate the assembly of personnel. The public address system will be used to evacuate specific areas. Upon hearing the emergency signal, all persons in the plant will go to their preassigned areas to wait completion of radiological surveys and further instructions. If only a specific area is to be evacuated, personnel in that area will evacuate to a safe area. Employees will be released from their assembly points when the Site Emergency Director determines it is suitable. Should evacuation of unnecessary site personnel be considered, routes, specific radiological conditions, traffic density, and weather conditions will be evaluated prior to directing their evaluation. Any necessary personnel decontamination identified during area evacuation and accountability will be accomplished prior to evacuation from the site.

13.3.5.2 Offsite

Through the assessment actions of the MSECC, the actual or potential offsite environmental conditions will be known. The State and local agencies will be responsible for implementing actions to protect the health and safety of the public. TVA recommends protective actions to these agencies but the State and local governments are responsible for deciding if any actions are needed and what they should be. TVA will assist State and local governments as necessary to implement protective actions for the public. TVA will also provide a prompt notification system for State and local governments to alert the public within a 10-mile area around the plant that protective actions may be required.

13.3.6 Review and Updating

The CRBRP-REP will be reviewed annually by the Division of Nuclear Power and the Division of Occupational Health and Safety for accuracy, completeness, operational readiness, and compliance with existing regulations. All holders of these plans will acknowledge in writing, receipt of all changes.

13.3.7 Medical Support

The CRBRP-REP will include a description of medical facilities at the plant and arrangements made with other facilities to provide additional support. One ambulance will be maintained at the site. Medical consultation will be available from TVA doctors in Chattanooga and other areas. Members of the plant emergency teams will be trained in first aid.

Arrangements will be made with a local hospital and with attending physicians for the emergency treatment of contaminated, injured, and exposed individuals. The Oak Ridge Associated Universities REAC/TS has agreed to provide treatment to severely contaminated or exposed individuals.

Arrangements will be made with a local private ambulance service to provide emergency service as required to the plant and affected areas in the event that more than one ambulance is required.

Figure 13.3-2 depicts the proposed locations of the Technical Support Center, the Operations Support Center, and the layout of the medical facilities and the personnel decontamination facilities.

13.3.8 Exercises and Drills

An exercise will be conducted at Clinch River Breeder Reactor Plant prior to the issuance of the full power operating license to test the CRBRP-REP, State, and local plans. Exercises will be conducted yearly thereafter. Drills will be conducted in the following areas:

1. Medical Emergencies
2. Radiological Monitoring
3. Radiochemistry
4. Transportation
5. Radiological Dose Assessment
6. Fire

13.3.9 Training

TVA will provide General-REP training to all plant personnel and specific training to emergency response personnel. This training provides emergency staff personnel with general knowledge of the emergency plan. Training on specific duties and emergency responsibilities will be provided personnel as necessary. This training will be such that each of these individuals will have a working knowledge of the emergency plan and his responsibilities and actions upon declaration of an emergency.

Training and periodic retaining will be provided to those offsite agencies who may be involved during an emergency, and will include procedures for notification, basic radiation protection, their expected roles, and site access procedures, as applicable. The Division of Occupational Health and Safety will provide for training to fire, police, ambulance, and hospital personnel from agencies with which TVA has agreement letters.

13.3.10 Recovery and Reentry

The CRBRP-REP will provide for the development and implementation of detailed recovery and reentry plans based on evaluation of conditions existing at the time. Recovery and reentry will be a deliberate, thoroughly planned evaluation and all procedures developed will be reviewed by the Plant Operations Review Committee prior to implementation.

13.3.11 Implementation

Operating instructions promulgated in the plant operating manual will be used to control plant operations during normal operating conditions. Abnormal operating instructions and emergency operating instructions which are contained in the Plant Operating Manual will be used to specify the manipulation of controls of the plant during conditions requiring protective measures to be taken to place the plant in a safe condition. The abnormal and emergency instructions will contain assignments of responsibility for the performance of specific tasks not otherwise established by plant practices and instructions.

Plant instrumentation indications requiring implementation of emergency and abnormal operating instructions will be specified in these instructions. Emergency action levels, also based on plant instrumentation indication, requiring implementation of the CRBRP-REP for protection of personnel and the environment are specified in the emergency plan.

Specific actions required of offsite TVA support groups will be delineated in the CRBRP-REP.

Instructions for medical treatment and handling of contaminated and exposed individuals will be contained in the CRBRP-REP.

Equipment requirements, including communications equipment, for implementation of the emergency plan will be contained in CRBRP-REP. Storage and calibration requirements will be specified. Alarm signals will be described in the plant procedures.

Instructions for restoring the emergency situation to normal, from the standpoint of the hazard to personnel, plant safety, and the environment, will be contained in the CRBRP-REP and the emergency and abnormal operating instructions. Instructions for repair of plant equipment or structures will be prepared after evaluation of the damage or malfunction involved.

TABLE 13.3-1

PARTICIPANTS IN CRBRP RAIDOLOGICAL EMERGENCY PLAN

Tennessee Emergency Management Agency

Tennessee Department of Public Health

Tennessee Department of Agriculture

Tennessee Department of Public Welfare

Tennessee Department of Safety

Tennessee Department of Conservation

Tennessee National Guard

Tennessee Game and Fish Commission

Tennessee Department of Transportation

City and County Officials of Roane, Anderson, Loudon, Morgan, and Knox Counties

Sheriff's Department of Roane, Anderson, Loudon, Morgan, and Knox Counties

Civil Defense Coordinators of Roane, Anderson, Loudon, Morgan, and Knox Counties

Local Police Departments

Local Ambulance Service

Local Fire Department

Radiological Emergency Assistance Center Training site (REAC/TS)

Oak Ridge Hospital of the United Methodist Church (ORHUMC)

Department of Energy (DOE)

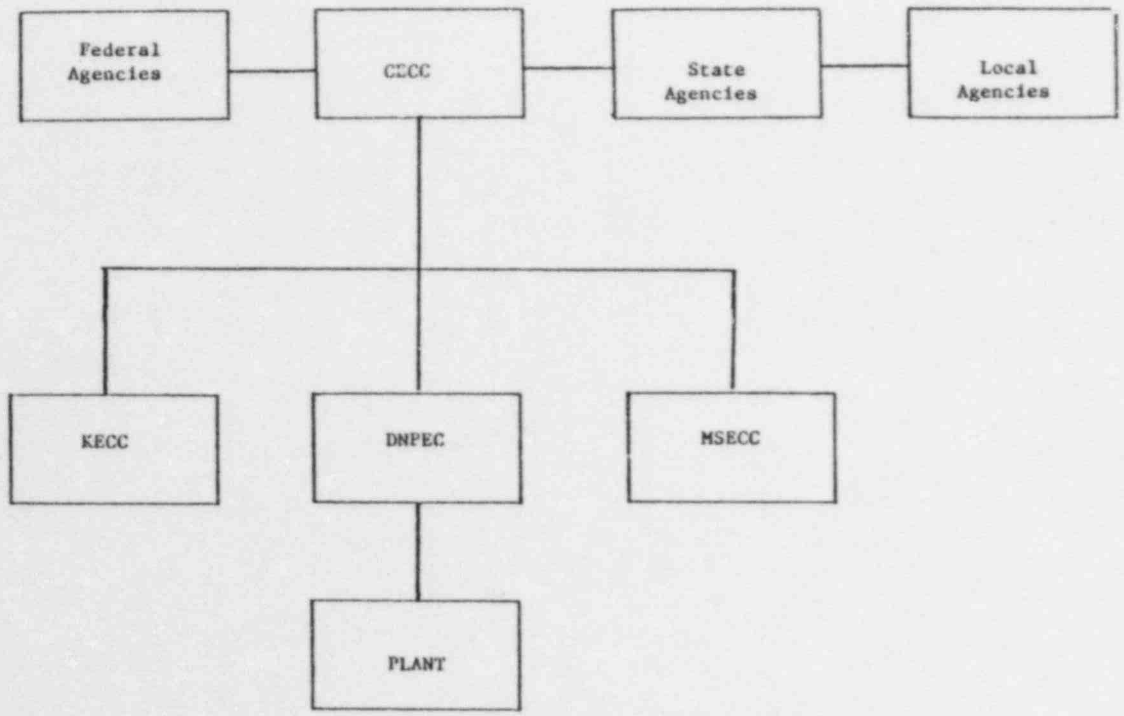
National Aeronautics and Space Administration

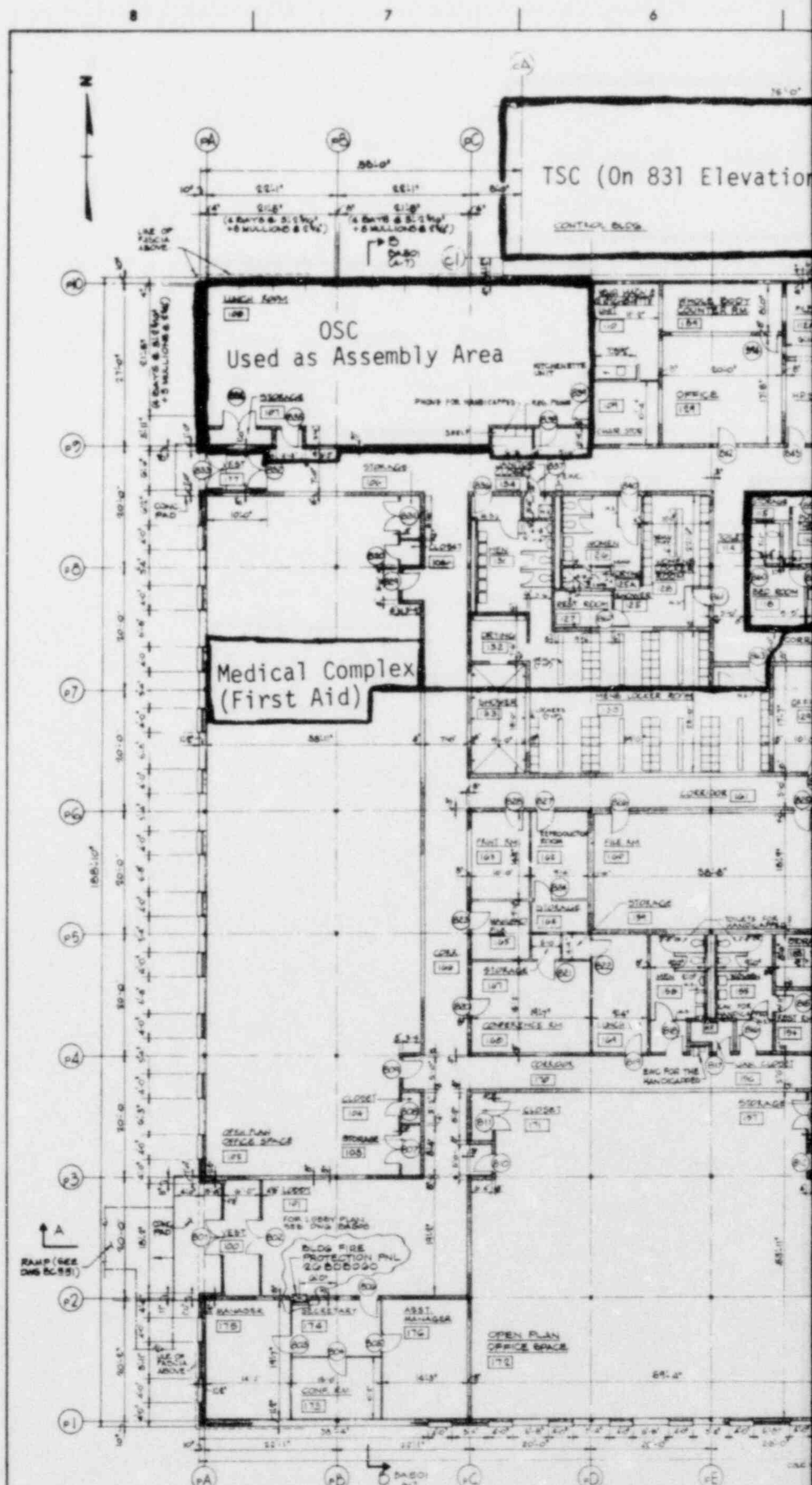
Nuclear Regulatory Commission

Environmental Protection Agency

FIGURE 13.3-1

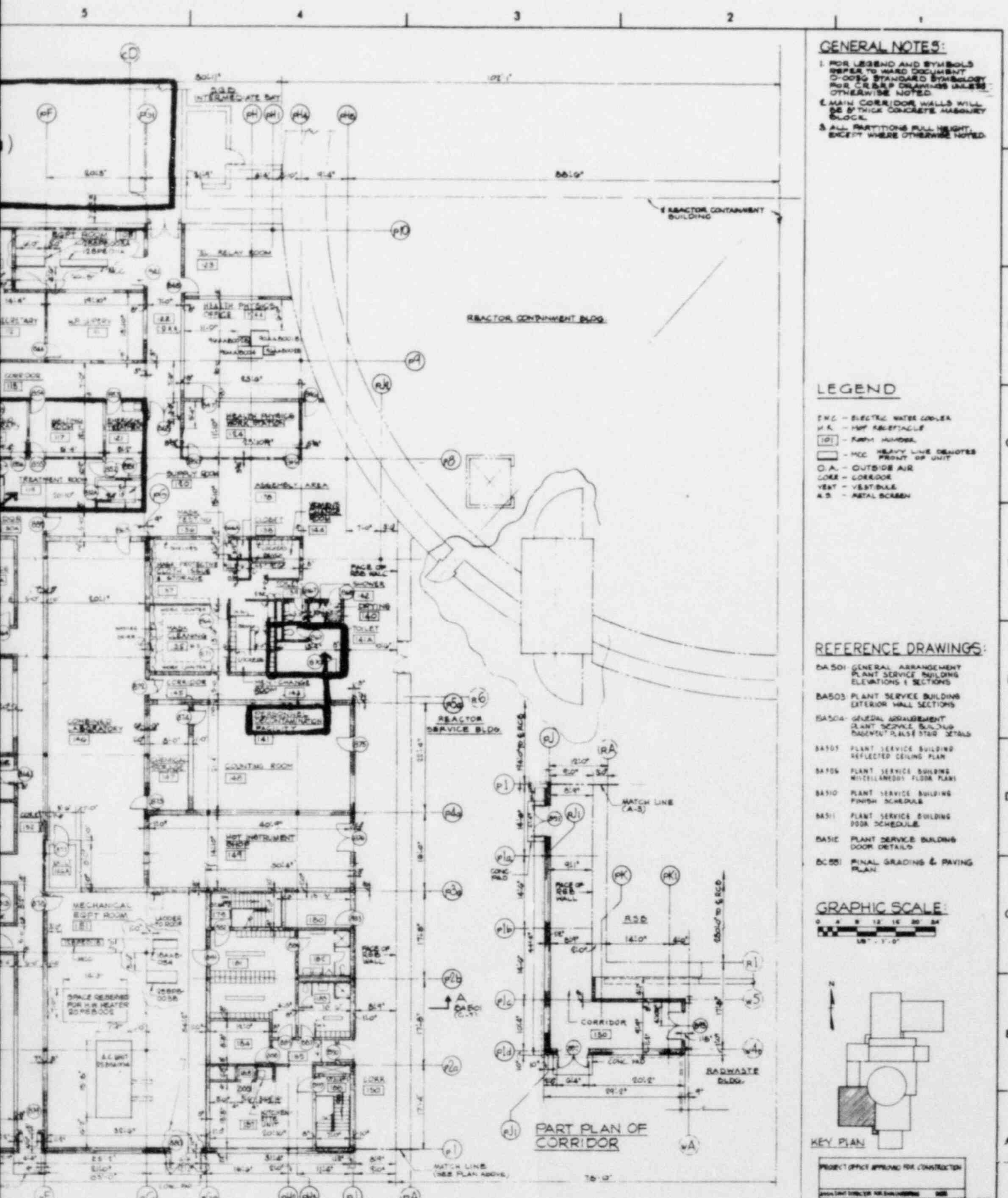
Relationship of Emergency Centers and Agencies





BA500-7

Locations of Technical
Support Center, Medica
Decontamination Facili



Support Center, Operational
l Facilities and Personnel
ties.

Figure 13.3-2

Amend. 69
July 1982

A. Introduction

A radiological analysis of the facility design features, site layout, and site location with respect to considerations of surroundings in compliance to 10 CFR 50, Appendix E-1982 has been conducted. The findings of this analysis are listed in this section.

B. Description of Analysis for Clinch River Project Emergency Planning Curves

Plots showing projected ground level doses, for both whole body and thyroid, resulting from the most serious design basis accident analysis is depicted in Figures 13.3A-1 through 13.3A-4. These provide, respectively, the elapsed exposure times to reach specific bone, lung, thyroid, and whole body doses as a function of downwind distance based on exposures resulting from the Site Suitability Source Term (SSST). The use of SSST is conservative since it envelopes the most serious design basis accident analyzed in the PSAR.

Source Term

100% Noble Gases

25% Halogens (50% release to containments, 1/2 of which [25% total] is airborne and available for release)

1% Solid Fission Products

1% Plutonium

Released instantly to and uniformly distributed in Reactor Containment Building (RCB).

Meteorology

Atmospheric dispersion parameters (x/Q's) are the ninety-fifth percentile values (see Section 2.3). Consistent with Regulatory Standard Review Plan, Section 2.3.4, the 0-2 hour exposure intervals were evaluated based on the single-hour 95% x/Q value.

Plume front transit times to downwind positions are based on a wind speed of 1 mile/hour.

Containment Modeling

The following parameters are used to evaluate Source Term releases from containment:

RCB Leakage to Annulus (Direct to Annulus Filter Intake)	0.1% Volume/Day

Annulus Flow Rates	
Filtered Exhaust	3000 CFM
Filtered Recirculation	3500 CFM per 1000 CFM Exhausted
Time Delay from Source Term Release to Initiation of Annulus Filtration	No Delay
Time Delay from Source Term Release to Initiation of Annulus Recirculation	<10 Seconds
Total Bypass Leakage (1% of RCB Leakage)	0.001% Volume/Day
Bypass Leakage Direct to Environment (60% of Total Bypass)	0.0006% Volume/Day
Bypass Leakage to the Reactor Service Building (RSB) (40% of Total Bypass)	0.0004% Volume/Day
Sources of Bypass Leakage to the RSB	96.4% Personnel and Equipment Airlock 3.6% All other Sources
Gamma Shielding	1.5" Steel (RCB) Plus 4' Concrete

Filter Efficiencies

Iodine	95%
Particulate	99%
Noble Gas	-0-

Radiological Parameters

Inhalation dose factors are per Regulatory Guide 1.109-R1-1977 for a standard adult.

Time dependent breathing rates are per Regulatory Guide 1.4-R2-1974.

External gamma whole body exposure is based on a semi-infinite cloud per Regulatory Guide 1.4-R2-1974 for the released material and includes direct exposure from the material within the Reactor Containment Building.

Radioactive decay of nuclides during downwind transit of the plume is conservatively neglected. While conservative, this assumption has minimal impact on the results, since off-site exposures are controlled by relatively long-lived nuclides.

C. Estimated Reaction and Response Times

1. The time required for the initial accident assessment of the most serious design basis accident may require 15 minutes. This time is an estimate based on the operation of the reactor instrumentation used to follow the course of accidents. Based on TVA's experience, the time required to perform an initial dose projection and notify offsite authorities can be accomplished in 15 minutes.

For the most serious design basis accident, the projected two-hour doses at the exclusion area boundary do not reach the protective action guide level for evacuation.

2. The time required to warn all resident and transient persons in any evacuation sector will conform to the requirements of 10 CFR 50, Appendix E-1982.

3. The estimated elapsed time, after the initial warning, to evacuate the 2-mile emergency planning zone (EPZ) is 4 hours. The estimated evacuation time for the 5-mile EPZ is 5 hours, 20 minutes. The estimated evacuation time of the 10-mile EPZ is 7 hours 15 minutes. Each estimate contains a 1-hour 50-minute preparation time factor.

4. These evacuation time estimates were prepared by the Traffic Management Division of the Tennessee Department of Transportation.

   a. Figures 13.3A-5 and 13.3A-6 are maps showing all roads within 10 miles of the Clinch River Project. Also indicated are the 2-, 5-, and 10-mile EPZ.

   b. Table 13.3A-1 shows the transient and resident populations in the 16 directional sectors within 10 miles of the Clinch River Project. This table uses 1980 census data.

   c. Table 13.3A-2 shows the estimated transient and resident populations in the 16 directional sectors within 10 miles of the Clinch River Project. This table uses the projected population figures for year 2020. The projected population figures come from a report prepared by the Firm of Dames and Moore dated June 16, 1981.

   d. Private automobiles will be the primary means for evacuating the population. Buses may be used to evacuate the Edgewood School. The problem will be specifically addressed in the CRBRP-REP.

5. Table 13.3-1 gives the agencies involved in the CRBRP emergency plan.

TABLE 13.3A-1
MAXIMUM RESIDENT AND TRANSIENT
POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 1980

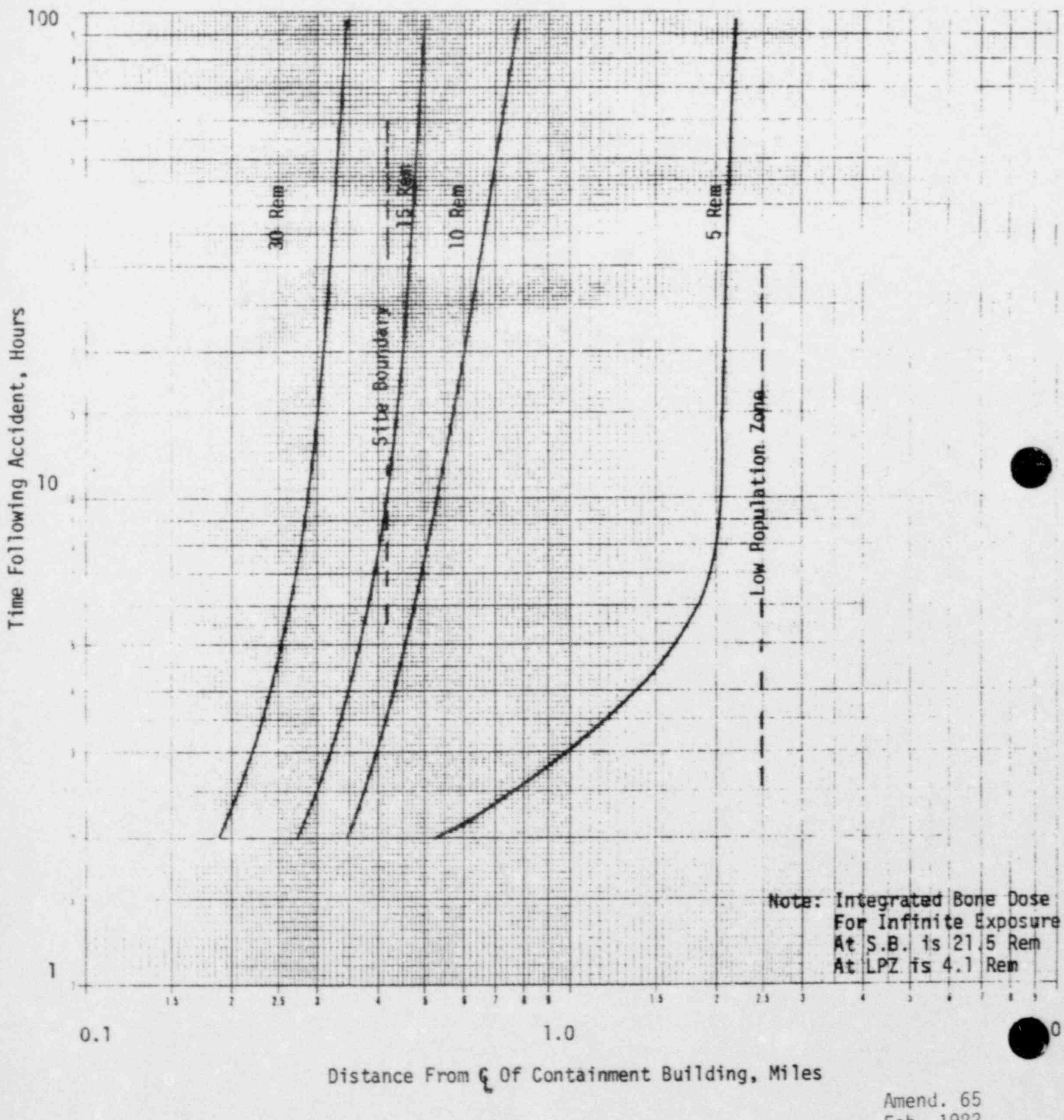
	Radial Interval (miles)					
Sector Designation	0-1	1-2	2-3	3-4	4-5	5-10
N	0	0	14	184	0	2,000
NNE	0	0	0	0	22	4,400
NE	0	0	0	8	80	7,191
ENE	10	10	0	8	0	4,728
E	20	30	50	398	3,568	5,172
ESE	20	30	50	187	159	2,300
SE	0	59	50	460	110	7,200
SSE	0	300	79	90	320	2,000
S	0	89	50	120	160	1,120
SSW	10	69	50	80	90	936
SW	20	80	119	110	140	1,292
WSW	20	70	80	193	340	5,000
W	0	130	114	110	991	6,764
WNW	10	94	170	10	60	4,676
NW	30	44	0	10	40	3,972
NNW	10	514	316	850	120	1,100
Sum for Radial Interval	150	1,519	1,142	2,818	6,200	59,851
Accumulative Total up to Radius Indicated	150	1,669	2,811	5,629	11,827	71,680

TABLE 13.3A-2
MAXIMUM RESIDENT AND TRANSIENT
POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 2020

Sector Designation	Radial Interval (miles) 0-1	1-2	2-3	3-4	4-5	5-10
N	0	0	14	184	0	2,000
NNE	0	0	0	0	22	8,300
NE	0	0	0	8	132	8,591
ENE	20	20	0	8	0	6,352
E	50	80	140	678	5,236	4,896
ESE	20	30	70	217	189	1,500
SE	0	69	70	758	70	11,500
SSE	0	492	89	110	210	1,300
S	0	109	60	150	200	936
SSW	10	79	70	100	110	1,108
SW	30	100	139	130	170	1,700
WSW	20	80	100	223	410	8,852
W	0	150	134	130	1,399	10,524
WNW	10	114	210	10	50	3,860
NW	30	44	0	10	40	5,032
NNW	10	774	424	850	100	900
Sum for Radial Interval	200	2,141	1,520	3,566	8,338	77,451
Accumulative Total up to Radius Indicated	200	2,341	3,861	7,427	15,765	93,216

FIGURE 13.3A-1

ELAPSED EXPOSURE TIME TO REACH SPECIFIC BONE DOSE VERSUS DOWNWIND DISTANCE (BASED ON SITE SUITABILITY SOURCE TERM)



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Feb. 1982

LEGEND

Primary highway, hard surface, heavy duty (24 ft.)
Secondary highway, hard surface, medium duty (24 ft.)
Light duty road, hard or improved surface (16-18 ft.)
Unimproved road
Road under construction, alinement known
Proposed road
Dual highway, dividing strip 25 feet or less
Dual highway, dividing strip exceeding 25 feet
Trail
Interstate Route U. S. Route State Route
Railroad: single track and multiple track
Bridge: road and railroad
Drawbridge: road and railroad
Footbridge
Tunnel: road and railroad
Barricaded road

Buildings (dwelling, place of employment, etc.)
School, church, and cemetery Cem
Buildings (barn, warehouse, etc.)
Power transmission line with located metal tower
Telephone line, pipeline, etc. (labeled as to type)
Wells other than water (labeled as to type) Oil Gas
Tanks: oil, water, etc. (labeled only if water) Water
Horizontal and vertical control station:
Tablet, spirit level elevation BM△5653
Other recoverable mark, spirit level elevation △5455
Horizontal control station: tablet, vertical angle elevation VABM△8
Any recoverable mark, vertical angle or checked elevation △3775
Vertical control station: tablet, spirit level elevation BM×957
Other recoverable mark, spirit level elevation ×954
Spot elevation

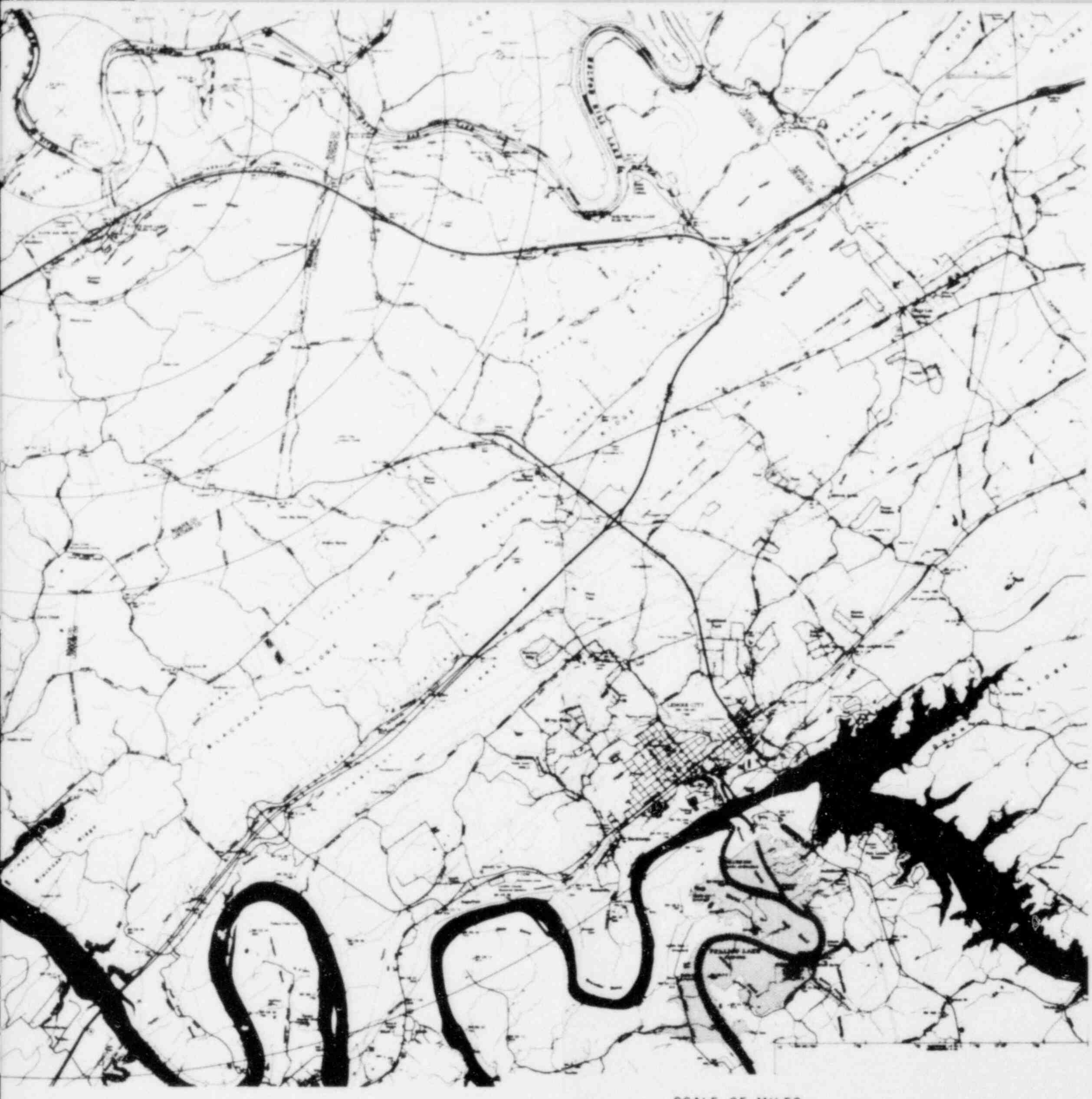
CRBRP MAP INF

NOTES

1. Four-Lane Road fr
2. 3 lane road: 1 lane reversib
3. 5 lane road: 3 lanes reversib
4. 3 lane road: 1 lane reversib
5. Guard at gate during day -
Bull Bluff Road is access to
6. This Bridge closed
7. one way bridge
8. Ferry

T-T Indicates - Includes portion
ref. notes 1, 2, 3, and 4 abo

See map legend for road widt



ORMATION

om JRCDP to intersection with White Wing Road

with directional signal lights

with directional signal lights

with directional signal lights

te locked at night

ecreational area

for an indefinite period

roads of more than 2 lanes

SCALE OF MILES

1 0 1 2

FIGURE 13.3A-6 Project Area - South Half

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Aug. 1982

13.4 REVIEW AND AUDIT

13.4.1 Review and Audit - Construction

The review and audit function during plant design and construction will be accomplished as an integral part of the quality assurance program described in Chapter 17.

13.4.2 Review and Audit - Test and Operation

This will be included in the FSAR.

13.5 PLANT PROCEDURES

13.5.1 Administrative Procedures

The plant administrative procedures will be provided at the FSAR stage. At that time both general and initial test program administrative procedures will be addressed.

13.5.2 Operating and Maintenance Procedures

Plant operating and maintenance procedures/instructions will be prepared for integrated plant operations, system operation, and plant maintenance activities following the generic categories listed in Table 3.5-1. They will be prepared by the plant operating personnel in accordance with NRC Regulatory Guide 1.33-R-2-1978. These procedures/instructions will be followed by the licensed operators in the operating the plant.

The preliminary schedule for the preparation of the operating and maintenance procedures/instructions is given in Table 13.5-1.

Table 13.5-1

Preliminary Schedule for O&M Procedures/Instructions

Title	Schedule for Preparation
General Operating Instructions	To be completed and approved at least six months prior to fuel loading and additional instructions developed as plant operation proceeds.
Normal Operating Instructions	To be completed and approved at least six months prior to fuel loading and additional instructions developed as plant operation proceeds.
Emergency Operating Instructions	To be completed and approved at least six months prior to fuel loading and additional instructions developed as plant operation proceeds.
Abnormal Operating Instructions	To be completed and approved at least six months prior to fuel loading and additional instructions developed as plant operation proceeds.
Maintenance Operating Instructions	To be completed and approved at least six months prior to fuel loading and additional instructions developed as plant operation proceeds.
Other Manuals/Procedures/Plans Fire Brigade Handbook Health Physics Manual Administrative Release Manual Division Procedure Manual Operational Quality Assurance Manual TVA Hazards Control Manual Nuclear Materials Management Guide Radiological Emergency Plan	To be complete and available at least six months prior to fuel loading.

FIGURE 13.5-1

DELETED

13.6 PLANT RECORDS

13.6.1 Plant History

The CRBRP operating record program, under TVA's responsibility as the plant operator, will observe all acts of Congress, Executive orders, and regulations of Federal agencies having jurisdiction in records administration and comply with 10 CFR 50, Appendix B, Section XVII-1982. TVA complies with Federal Power Commission regulations concerning the preservation and disposal of records of public utilities and licensees, insofar as these regulations apply to TVA records relating to the generation, transmission, and sale of electric energy.

The Plant Administrative Officer will have responsibility for the general supervision and coordination of the plant master file.

13.6.2 Operating Records

This will be included in the FSAR.

13.6.3 Event Records

This will be included in the FSAR.

13.7 RADIOLOGICAL SECURITY

The requirements of applicable provisions of: 10 CFR Part 11-1982; Part 25-1982; Part 50 [these applicable Paragraphs, 34(c), 34(d), 34(e)-1981-82]; Part 70 [Paragraph 20a-1980]; Part 73 [these applicable Sections, 20, 21, 25, 26, 45, 46, 55, 71, 72, 80,-1980-82 Appendices B-1981 and C-1978]; and Part 95-1980-82 will be followed at CRBRP. This section discusses in general how the CRBRP will meet the requirements and each of the Parts and/or Paragraphs listed above will be discussed at the appropriate paragraph. The CRBRP Physical Security Plan, Safeguards Contingency Plan, and Security Personnel Training and Qualification Plan will be submitted as separate safeguards information documents. Specific details will be provided by these plans.

13.7.1 Organization and Personnel

The Office of Power of the Tennessee Valley Authority (TVA) will have the security responsibilities for the CRBRP for the period from receipt of fuel to the end of operation. The organization chart shown in Figure 13.7-1 delineates this responsibility which is explained in Sections 13.7.1.1 and 13.7.1.2.

13.7.1.1 Office of Power

The Office of Power is responsible for protection of power properties. It develops detailed plans and applies specific security measures.

The Division of Nuclear Power is responsible for security of operating nuclear plants. The Nucler Power Security Section prepares required physical security plans and Safeguards Contingency Plans for approval by the Manager of Power. The Director, Division of Nuclear Power, approves the Guard Training and Qualification Plan.

13.7.1.2 Office of Management Services

The security force at the CRBRP will be provided by the Public Safety Service (PSS) Branch in the Office of Management Services of the Tennessee Valley Authority. The Public Safety security unit at CRBRP will be under the administrative supervision of the supervisor of the Nuclear Operations Section, located in the Public Safety Service Branch office, but he will be under the functional supervision of the plant manager.

13.7.1.3 Employee Selection

As discussed in Section 13.1, TVA will operate the plant, and accordingly, will provide all operating and security personnel who will be regular TVA employees.

TVA appoints, promotes, transfers, and retains employees on the basis of merit and efficiency, as prescribed in the TVA Act and in accordance with other applicable Federal laws and regulations. It is the policy of TVA to promote present employees, whenever possible, who have demonstrated competence, reliability, and stability to vacant positions in preference to hiring persons from outside the organization. This is often accomplished by upgrading employees through internal training programs.

Specific instructions pertaining to personnel matters are contained in Section III of the TVA Administrative Release Manual. These instructions are observed by all plant supervisors, especially as they apply to appointment, transfer, promotion, and retention of employees.

Selection for a position is supportable by records of education, training, and experience, and by records of judgements which have been made regarding work performance, ability, and condition of health.

In selecting for placement or retention in positions, covered by agreements negotiated between TVA and the employee organizations, the provisions of such agreements are observed.

Because of TVA's conformance to the Veteran's Preference Act, when employing outside candidates for vacant positions, a large number of persons beginning employment have successfully completed tours of duty with the military forces of the USA. The availability for review of the military record of these candidates provides good control in the selection of high-quality candidates.

Each new TVA employee is given a physical examination and a national agency check, and written inquiries are routinely made to references such as former employers, schools, and police. Before any employee is allowed unescorted access to a nuclear plant protected area, there must be satisfactory results from his security check and emotional stability check.

Public Safety (PS) officer selection procedures include a preemployment interview by the PSS area chief and one or more PSS unit supervisors in addition to the steps previously mentioned. Upon acceptance, the candidate's first six months of employment are probationary. Appointment as a PS officer is dependent upon satisfactory service during this period and satisfactory completion of training and qualification as provided for in the CRBRP Training and Qualification Plan to be submitted with the CRBRP Physical Security Plan.

In the course of hiring personnel, the spirit and intent of 10 CFR Parts 11-1982 and 25-1982 will be adhered to in considering eligibility for Special Nuclear Material access and access to National Security information.

13.7.1.4 Employee Evaluation

Because of the general policy of promoting present employees rather than appointing candidates from outside TVA, most employees at the CRBRP will be known from their previous employment record with TVA. Although TVA employees are not given routine psychiatric examinations, they will be given when an employee's on-the-job performance indicates that this is desirable.

Observation of employee service is made as a regular part of day-to-day continuous supervisory function. When performing this function, supervisors will be alert for any unusual behavioral patterns such as may result from mental stress, alcohol, or other drug abuse.

In addition to this kind of review, the performance of employees in management and salary policy positions are reviewed formally and the results reported in order (1) to further aid in maintaining a high level of employee performance and the maximum utilization of employee abilities; (2) to provide recorded evidence of employee performance for use in making judgements concerning transfer, demotion, promotion, and terminations; (3) to assure that employees are adequately and systematically informed of the effectiveness of their service; and (4) to further facilitate the maintenance of a high standard of supervision in TVA. All employees' services are reviewed formally at the time of status changes and at such other times as may be required to achieve the above purposes. A service review shall precede each recommendation for operator licensing or renewal of an operator license.

13.7.1.5 Industrial Security Training

All employees at CRBRP will receive training in security procedures with emphasis on being alert to the presence of unauthorized persons and evidence of forced entry. This training will normally be conducted by a member of the Plant Security Force under the direction of the Plant Manager. The security force training and qualification will be in accordance with CFR 10 Part 73, Appendix B-1981.

13.7.2 Plant Design

The physical plant design has been developed so as to accommodate the necessary security provisions. TVA, along with DOE and PMC and its architect-engineer, Burns and Roe, will provide a continuing review of the plant design, as well as the detailed security provisions. Burns and Roe, as the architect-engineer for the Project, has been delegated the responsibility for detailing the security provisions. The design criteria used at the CRBRP will assure that the physical security facilities and the plant layout will be developed so as to thwart any attempted sabotage. All plans, drawings, design information, etc., related to the security system will be treated in accordance with 10 CFR Section 73.21-1981 or 10 CFR Part 95-1980-82 as appropriate. The physical security design will:

(1) Control entry to the plant site and portions of the plant;

(2) Deter or discourage penetration by unauthorized persons;

(3) Detect such penetrations in the event they occur; and

(4) Apprehend in a timely manner unauthorized persons or authorized persons acting in a manner constituting a threat of sabotage.

In the design and operation of the plant, care will be taken to minimize the potential for radiological sabotage by the use of access control measures to prevent unauthorized persons from entering the protected area. Should such persons succeed in entering the protected area, special access control measures will prevent them from entering vital equipment areas and the Special Nuclear Material (SNM) material access area.

All of the design features will be in accordance with the requirements and the performance objectives of 10 CFR Sections 73.20-1981 and 73.55-1980. In those cases where SNM is present, the additional requirements of 10 CFR Sections 73.45-1982 and 73.46-1980 will be met. The design features will meet the performance objective with a high degree of assurance based on detailed analysis.

13.7.2.1 Design Features

The design features of this plant will meet the performance objectives and requirements of 10 CFR 73.20-1981 in that it will provide with a high degree of assurance that activities involving special nuclear material will not be inimical to the common defense and security, and will not constitute an unreasonable risk to the public health and safety. The physical protection system will be designed to protect against the design basis threats of theft or diversion of strategic special nuclear material and radiological sabotage as stated in paragraph 10 CFR 73.1(a)-1982. Design features will be:

a. A security barrier with intrusion detection system around the perimeter of the plant. Gates will be kept closed and locked except during times of authorized use.

b. Employee and visitor parking located outside the security barrier.

c. An isolation zone extending from inside the security barrier to outside the barrier in which all activities will be controlled. This zone will be void of obtrusive structures and plant growth. In addition, a cleared zone will be maintained outside the isolation zone to facilitate observation of persons approaching the isolation zone.

d. A patrol road extending completely around the plant inside the security barrier.

e. Outdoor closed circuit television (CCTV) systems to permit observation of the plant perimeter, isolation zone, cleared zone, and protected area.

f. An outdoor lighting system to provide illumination to the protected area and isolation zone at a level compatible for both visual and CCTV observation.

g. A minimum number of exterior plant doors leading to vital areas, all of which will be hardened against penetration and kept locked.

h. A cardkey electronic access control system to control personnel access to vital areas in conformance with each employee's level of authorization.

i. An intrusion detection system to indicate status of hatches, emergency exits and seldom used equipment or personnel access doors providing access to vital areas.

j. An access control facility to control personnel access to the protected area and containing equipment to search personnel for weapons and explosives.

k. A communication system which will allow continuous communications between PSS officers and the central alarm station. Also, redundant communications links will be maintained between the plant and the local law enforcement agency.

l. An electric power system to provide emergency power to the security and lighting loads.

m. A force of trained, uniformed, and armed PS officers used on a three-shift basis to police the property, provide access control, respond to alarms, evaluate the situations, and neutralize the threats. Guards will be trained to meet the requirements of 10 CFR Part 73 Appendix B-1981.

n. Firefighting and other emergency equipment located throughout the plant area to minimize the consequences of fires or explosions.

o. Engineered safeguards and protective systems that will be provided to minimize the consequences of fires or explosions or to minimize the effects of postulated major equipment failures, natural disasters, and operator errors which would also serve to minimize the effects of radiological sabotage.

13.7.2.2 Physical Arrangements

The CRBRP site is in a remote location. It is unlikely that major civil disorders would occur at or near the plant area. The plant will be located on a peninsula formed by a meander of the Clinch River between river miles 14.5 and 18.6 near the center of a 1364-acre tract owned by and in the custody of the United States Government (see Section 2.1).

13.7.2.3 Owner-Controlled Area

Ultimately, a permanent access road to the plant will lead into the plant. During construction, a temporary construction road will lead into the construction area. The perimeter of the reservation will be marked prior to the completion of construction with signs to provide reasonable assurance that persons entering the area are aware they are on private property. Employee parking areas will be located outside the security barrier so that only plant vehicles and trucks making deliveries will need to be admitted. While construction is in progress, the temporary construction road will be the only route of access to the Project. A continuous access control guard post will be maintained on this road for the duration of construction activities. Section 2.1 shows the reservation boundary of the owner-controlled area.

13.7.2.4 Protected Area

When all construction work is completed, there will be an 8-foot high perimeter security barrier enclosing the protected area. An isolation zone will be maintained both outside and inside the security barrier which meets or exceeds the NRC requirement as specified in 10 CFR 73.55-1980. A patrol road will be located inside this barrier. A sectionalized instrusion detection system designed to be self-checked and tamper-indicating will be located along the barrier with sensors located on or between it and the patrol road. Closed-circuit television (CCTV) systems using lowlight level cameras including some with zoom lens and remote pan and tilt controls will be used to provide a means of promptly viewing the sector or general area involved. Safeguards Information Figure 13.7-2 indicates compliance with 10 CFR Sections 73.20, 45, 46 and 55,-1980-82, and are treated as safeguards information in accordance with 10 CFR Section 73.21-1981.

13.7.2.5 Vital Equipment and Vital Areas

All vital equipment and material access areas will be located within a vital area or building which, in turn, will be located within a protected area. Doors and gates to vital areas and to other selected sensitive areas will be kept closed and locked at all times. Safeguards Information Figures 13.7-6 through 13.7-11 indicate compliance with ANSI N18.17-1973, Section 3.4, and other applicable guides and regulations.

As construction nears completion and the equipment is made operational, the doors and gates to vital areas will be identified by signs which state that entry through them will be with the permission of the shift engineer on a need basis. Upon completion of construction, these doors and gates will be controlled by a cardkey access control system.

The cardkey system will be self-checking and tamper-indicating and an emergency power source provided. The regular power supply and emergency supply will be supervised and the operation of each cardkey controlled door tested no less frequently than once each seven days.

All issues of cardkeys will be authorized by the Plant Manager according to individual needs of employees requiring access to areas controlled by the cardkey system. Each card in the system will be programmed individually and can be programmed out at any time if lost. The cards will be issued and returned daily to insure that they do not leave the site.

13.7.2.6 Alarm Station

All intrusion detection devices will annunciate in a continuously manned central alarm station, and in a secondary alarm station, both of which will meet the requirements of CFR 10 Subparagraph 73.46(e)(5)-1980.

Each sector of the outdoor intrusion detection system and the operation of each cardkey controlled door will be tested no less frequently than once each seven days. Onsite and offsite communication facilities will be tested at the beginning of each security force workshift.

13.7.2.7 Security Barrier

The protected area barrier will consist of an 8-foot high No. 9 gauge chain-link fence (7-feet of fabric and 3 strands of barbed wire on angle brackets). The alignment of the security barrier will have a minimum number of angles and curves to facilitate effective observation and maximum length sectors of the intrusion detection system.

13.7.3 Security Plan and Contingency Plan

The Plant Physical Security Plan will describe security measures used to minimize the potential for industrial sabotage, theft and/or diversion including access control, surveillance of vital equipment, and plans for responding to security threats and will be written in accordance with 10 CFR Paragraph 50.34(c)-1981. The Contingency Plan, which will be written to be mutually supporting with the security plan, will be written in accordance with 10 CFR 50.34(d)-1982. Both plans will be treated as Safeguards Information in accordance with 10 CFR Subparagraph 73.21 (b)(1)-1981 and Part 95-1980-82 as appropriate.

13.7.3.1 Access Control

The CRBRP will have a perimeter security barrier that encloses all vital areas. The plant will have two portals for normal access: (1) A personnel portal and (2) a nearby vehicle gate. General public visitors will not be permitted inside the security barrier. Employees and special visitors' parking areas will be located outside the security barrier. Vehicle access will be limited to those required for delivery of material, operations, maintenance, and security of the plant. Persons, packages, and vehicles will be subject to search upon entering, leaving, and while within the plant area.

There will be a minimum number of exterior accesses to the CRBRP buildings. All of them will have penetration resistant doors with frames, hinges, and locks or security devices designed to prevent forced entry and will be alarmed or cardkey controlled. These doors will be kept locked.

All persons authorized to enter the protected area unescorted will have had a satisfactory security check and emotional stability check and will have completed as a minimum a brief plant indoctrination and radiation protection course which describes plant organization and layout, controlled zones, radiation and contamination hazards, exposure limits and controls, elementary health physics, and pertinent sections of the site emergency plan.

Even those persons who are authorized unescorted access will have their movement limited by physical barriers, such as locked doors, to prevent them from entering areas containing vital equipment or areas of high radiation levels. Only those who need access to these areas will be provided means of entering. Access to vital areas and material access areas will be controlled to meet the performance objectives of 10 CFR Parts 11-1981-82 and 25-1982, and Sections 73.20-1981 and 73.45-1982.

When special visitors and other persons who have not completed this training enter the protected area, they will be escorted by an employee trained in radiation protection and plant emergency procedures. The escort will be responsible for the safety and security of the people in his charge until they leave the plant protected area.

13.7.3.2 Control of Personnel by Categories

Employees and visitors authorized unescorted access to the plant protected areas by the Plant Manager will be issued a photo-type identification (ID) badge with tamper-resistant features. These persons will be identified, issued a radiation detection badge and dosimeters, and then be admitted to the protected area. Other persons not issued ID badges who require escorts may be identified by personal recognization, TVA identification card, or other available identification media. They will be issued a numbered visitor's badge to be worn on their person while within the plant protected area. Access to vital areas and material access areas will be in conformance with the Performance Objectives of 10 CFR Parts 11-1981-82 and 25-1982, and Sections 73.20-1981 and 73.45-1982.

The only unescorted construction workers who may be inside the confines of the operating unit will be those selected to perform maintenance or other work before final acceptance of equipment. Upon being granted unescorted access by the Plant Manager, these persons will be issued a photo ID badge and given a brief security indoctrination course covering the evacuation procedure and relevant sections of the site emergency plan.

Contractor personnel, manufacturers' representatives, and other special visitors who require access to the plant will be logged in, badged, and escorted.

13.7.3.3 Access Control During Emergencies

Upon hearing of an emergency, the access portal will be locked to ensure controlled entry and exit. Special visitors who are onsite will be escorted to the access portal.

Plant employees will report to the predesignated stations from which they will be dispatched as needed to combat the emergency. All access control procedures will be compatible with the CRBRP Radiological Emergency and Contingency Plans.

13.7.3.4 Surveillance of Vital Equipment and Material Access Areas

Unit operators will continuously monitor the status of plant systems and equipment by means of annunciators, indicating lights, indicators, and recorders. New equipment or material will be inspected on delivery. Operating logs and computer printout data will be periodically examined for changes in equipment performance. Most equipment will be in continuous operation and any change will immediately be detected by the operator. Standby and emergency equipment will be periodically tested on a routine basis as required by the technical specifications. Procedures will be employed to control access to the material access area in accordance with 10 CFR Part 11-1981-82 and Sections 73.45-1982 and 73.46-1980. In addition, activities in the vicinity of the material access area will be monitored. The combination of these efforts should provide reasonable assurance that unauthorized physical changes in the status of components of equipment will not be undetected for long periods of time.

Accountability of fissile and fertile material is inherent in the design of the CRBRP refueling system for reasons other than security. After inspection at receipt, the assemblies will not be visually identified again until shipment of the irradiated assemblies. The assemblies will be mechanically identified prior to insertion into the core and subsequent to removal from the core as part of the reactor safety program. All movements of fuel within the plant will be monitored and/or recorded on the refueling system computer for inventory purposes and to insure reactor safety during core configuration changes.

13.7.3.5 Potential Security Threats

The security system will be designed to deter unauthorized persons from entering the protected and vital areas and to detect such attempts. Operating personnel will be trained to be alert for unauthorized persons and to appropriately notify the security force.

Detailed descriptions of decisions/actions regarding potential security threats will be included in the CRBRP Contingency Plan. The Contingency Plan will address actions in the event of an attempt at plant break-in or an indication of an attempt of theft or diversion or an actual theft, diversion or sabotage. Planned actions in the event of civil disturbances, bomb threats, and other emergencies, will be included. Detailed procedures will be provided plant employees so that they may cope with these and other events in the optimum manner possible. The Contingency Plan will be protected in accordance with the requirements of 10 CFR Section 73.21-1981.

13.7.3.6 Administrative Procedures

In the event of an incident of suspected sabotage or condition which threatens the security of the plant, the security force will notify the

Plant Manager and initiate a thorough investigation. A report will be prepared which includes as a minimum the cause of the event, extent of damage, if any, and action taken to prevent recurrence of similar event. Copies of the report will be sent to the Plant Manager. When appropriate, the Plant Manager will also report the situation to NRC.

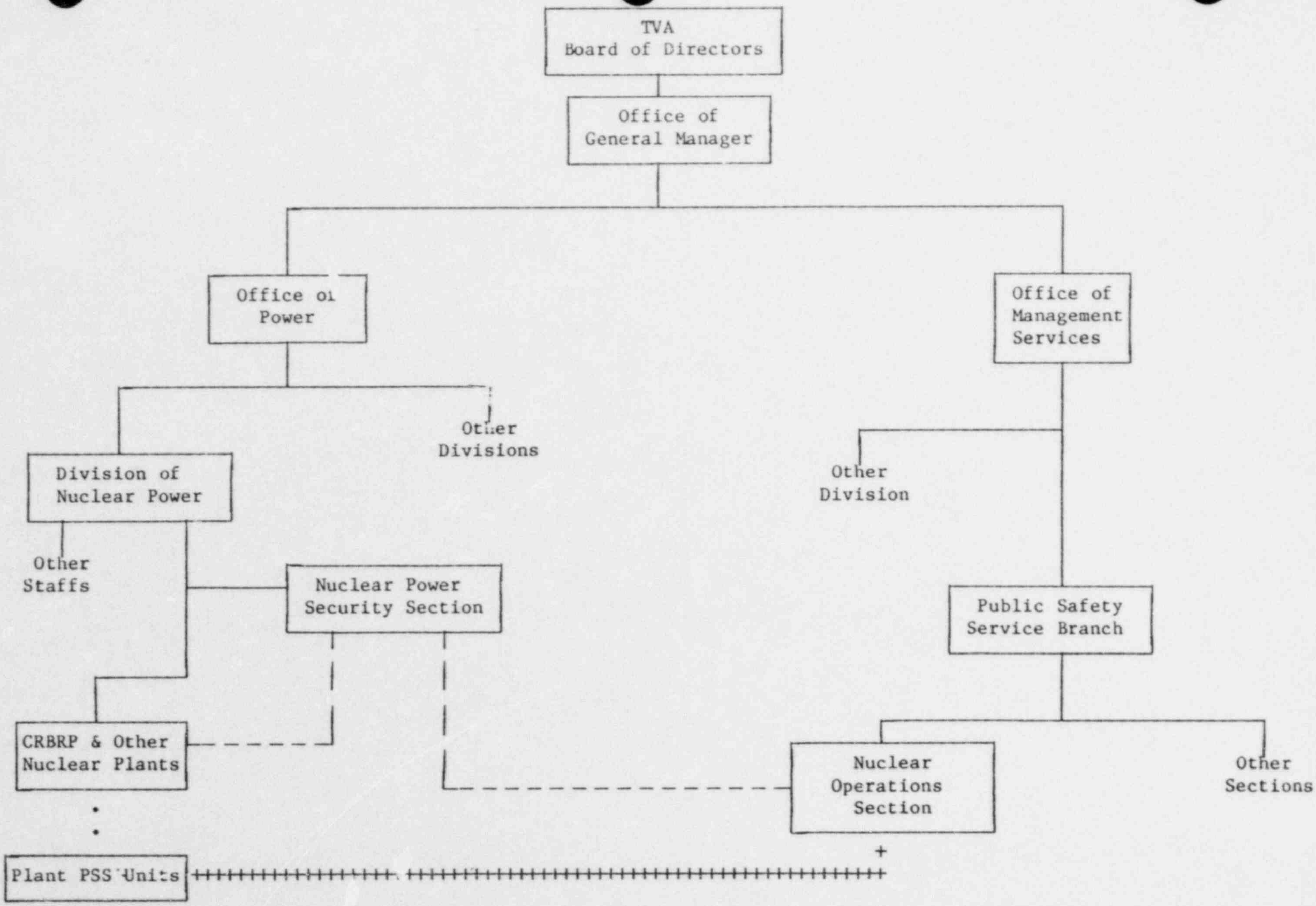
Representatives of the Nuclear Power Security Section and Office of Power Quality Assurance and Audit Staff will make an annual audit of the CRBRP Physical Security Program for adequacy of content and performance. Based on their audit, they will make recommendations for revising and updating the plant and related plant procedures.

13.7.3.7 Test and Inspections

This information will be supplied in the CRBRP Physical Security Plan.

13.7.4 Transportation of Fuel

The transportation of fuel for CRBRP will be the responsibility of the Department of Energy, both for delivery of new fuel and removal of spent fuel. Requirements virtually identical in objective and scope to those set forth in 10 CFR Sections 70.20-1981, 73.25-1981, and 73.26-1981 will be met. All plans concerning moves of fuel will be controlled in accordance with the appropriate directives of the agency in charge, i.e., DOE or NRC.



— — — Coordination
++++++Administrative Responsibility
••••••Functional Responsibility

Figure 13.7-1
CRBRP Security Organization

Figures 13.7-2 and 13.7-6 through 13.7-11
are considered as Safeguards Information
and are not included in the PSAR.

TABLE 15.1.3-2

ACCIDENT EVENTS

PSAR Section	Anticipated Events
15.2.1.1	Control Assembly Withdrawal at Startup
15.2.1.2	Control Assembly Withdrawal at Full Power
15.2.1.3	Seismic Reactivity Insertion (OBE)
15.2.1.4	Small Reactivity Insertions
15.2.1.5	Inadvertent Drop of a Single Control Rod at Full Power
15.3.1.1	Loss of Off-Site Electrical Power
15.3.1.2	Spurious Primary Pump Trip
15.3.1.3	Spurious Intermediate Pump Trip
15.3.1.4	Inadvertent Closure of One Evaporator or Superheater Module Isolation Valve
15.3.1.5	Turbine Trip
15.3.1.6	Loss of Normal Feedwater
15.3.1.7	Inadvertent Actuation of the Sodium/Water Reaction Pressure Relief System
15.7.1.1	Loss of One D.C. System
15.7.1.2	Loss of Instrument or Valve Air System
15.7.1.3	IHX Leak
15.7.1.4	Off-Normal Cover Gas Pressure in PHTS
15.7.1.5	Off-Normal Cover Gas Pressure in IHTS
	Unlikely Events
15.2.2.1	Loss of Hydraulic Holddown
15.2.2.2	Sudden Core Radial Movement
15.2.2.3	Maloperation of Reactor Plant Controller
15.3.2.1	Single Primary Pump Seizure
15.3.2.2	Single Intermediate Loop Pump Seizure
15.3.2.3	Small Water to Sodium Leaks in Steam Generator Tubes
15.3.2.4	Failure of Steam Bypass System
15.5.2.1	Fuel Assembly Dropped During Refueling
15.5.2.2	Attempt to Insert a Fuel Assembly into an Occupied Position
15.5.2.3	Single Fuel Assembly Cladding Failure in Fuel Handling System
15.5.2.4	Cover Gas Release During Refueling
15.5.2.5	Heaviest Crane Load Impacts Reactor Closure Head
15.7.2.1	Inadvertent Release of Oil Through the Pump Seal (PHTS)
15.7.2.2	Inadvertent Release of Oil Through the Pump Seal (IHTS)
15.7.2.3	Generator Breaker Failure to Open at Turbine Trip
15.7.2.4	Rupture in the RAPS Cold Box
15.7.2.5	Liquid Radwaste System Failure
15.7.2.6	Failure in the EVST NaK System
15.7.2.7	Leakage from Sodium Cold Traps
15.7.2.8	Rupture in RAPS Noble Gas Storage Vessel Cell
15.7.2.9	Rupture in the CAPS Cold Box

TABLE 15.1.3-2 (Continued)

Extremely Unlikely Events

15.2.3.1	Cold Sodium Insertion
15.2.3.2	Gas Bubble Passage Through Core, Radial Blanket and Control Assembly
15.2.3.3	Core, Radial Blanket, and Control Rod Movement Due to Safe Shutdown Earthquake
15.2.3.4	Control Assembly Withdrawal at Startup - Maximum Mechanical Speed
15.2.3.5	Control Assembly Withdrawal at Power - Maximum Mechanical Speed
15.3.3.1	Steam or Feed Line Pipe Break
15.3.3.2	Loss of Normal Shutdown Cooling System
15.3.3.3	Large Sodium/Water Reactions
15.3.3.4	Primary Heat Transport System Pipe Leak
15.3.3.5	Intermediate Heat Transport System Pipe Leak
15.5.3.1	Collision of EVTM with Control Rod Drive Mechanisms
15.6.1.1	Primary Sodium In-Containment Storage Tank Failure During Maintenance
15.6.1.2	Failure of the Ex-Vessel Storage Tank Sodium Cooling System During Operation
15.6.1.3	Failure of Ex-Containment Primary Sodium Storage Tank
15.6.1.4	Primary Heat Transport System Piping Leaks
15.6.1.5	Intermediate Heat Transport System Piping Leak
15.7.3.1	Leak in a Core Component Pot
15.7.3.2	Spent Fuel Shipping Cask Drop from Maximum Possible Height
15.7.3.3	Maximum Possible Conventional Fires, Flood or Storms or Minimum River Level
15.7.3.4	Failure of Plug Seals and Annuli
15.7.3.5	Fuel Rod Leakage Combined with IHX and Steam Generator Leakage
15.7.3.6	Sodium Interaction with Chilled Water
15.7.3.7	Sodium-Water Reduction in Large Component Cleaning Vessel

TABLE 15.1.4-3

PARAMETRIC CASES TO DETERMINE WORST CASE FOR NEUTRONIC POWER VARIATION DURING THE LOSS OF OFF-SITE ELECTRICAL POWER EVENT

CASE	CONDITIONS	$P/P_{o,n}$*
1 (Base Case)	o Minimum fuel C_p o Longest flow coastdown o Maximum Doppler o Zero decay heat o Maximum fuel/cladding gap conductance o Zero sodium coolant density feedback	0.1327
2	o Quickest flow coastdown o Other conditions same as Case 1	0.1322
3	o Maximum sodium coolant density feedback o Other conditions same as Case 1	0.1326
4	o Maximum decay heat o Other conditions same as Case 1	0.1322
5	o Maximum fuel C_p o Other conditions same as Case 1	0.1325
6	o Minimum fuel/cladding gap conductance o Other conditions same as Case 1	0.1304
7	o Minimum Doppler o Other conditions same as Case 1	0.1248

*P = Nuetronic power at 2 seconds (selected for comparison only) into the transient.

$P_{o,n}$ = Neutronic power at time 0 of the transient.

TABLE 15.1.4-4

TEMPERATURES BEHIND A CENTRAL SIX CHANNEL BLOCKAGE IN FUEL ASSEMBLIES

	PEAK PIN(3)	HOT PIN(4)
Maximum wake temperature increase, ^{o}F	270	116
Average wake temperature increase, ^{o}F	180	77
Maximum wake temperature, ^{o}F	1216	1385
Maximum cladding temperature, ^{o}F(1)	1367	1432
Dimensionless residence time, t_R(2)	22.5	22.5
Linear power rating, Kw/ft	14.1	4.76

---

(1) Based on maximum fluid temperature.

(2) $t_R = d_B \frac{\tau U}{}$ where: τ is the average residence time of the fluid in the wake region, U is the free stream velocity; and d_B is the characteristic blockage dimension. The temperature increase is proportional to t_R. The t_R value 22.5 used here is conservative.

(3) Blockage is conservatively assumed to occur at peak power spot, i.e., core Midplane.

(4) Blockage is conservatively assumed to occur at hot spot, i.e., top of the core.

TABLE 15.1.4-5

TEMPERATURES BEHIND A SIX-CHANNEL BLOCKAGE IN INNER BLANKET ASSEMBLIES

	PEAK PIN(3)	HOT PIN(4)
Maximum wake temperature increase, °F	462	167
Average wake temperature increase, °F	308	111
Maximum wake temperature, °F	1410	1385
Maximum cladding temperature, °F(1)	1526	1430
Linear power rating, KW/ft	18.4	6.7
Dimensionless residence time, t_R(2)	22.5	22.5

---

(1) Based on maximum fluid temperature.

(2) The value of t_R was calculated to be 17 (Section 15.4.3.3.3). For conservatism, the same t_R value 22.5 is used here.

(3) Blockage assumed to occur at core Midplane.

(4) Blockage assumed to occur at top of the core.

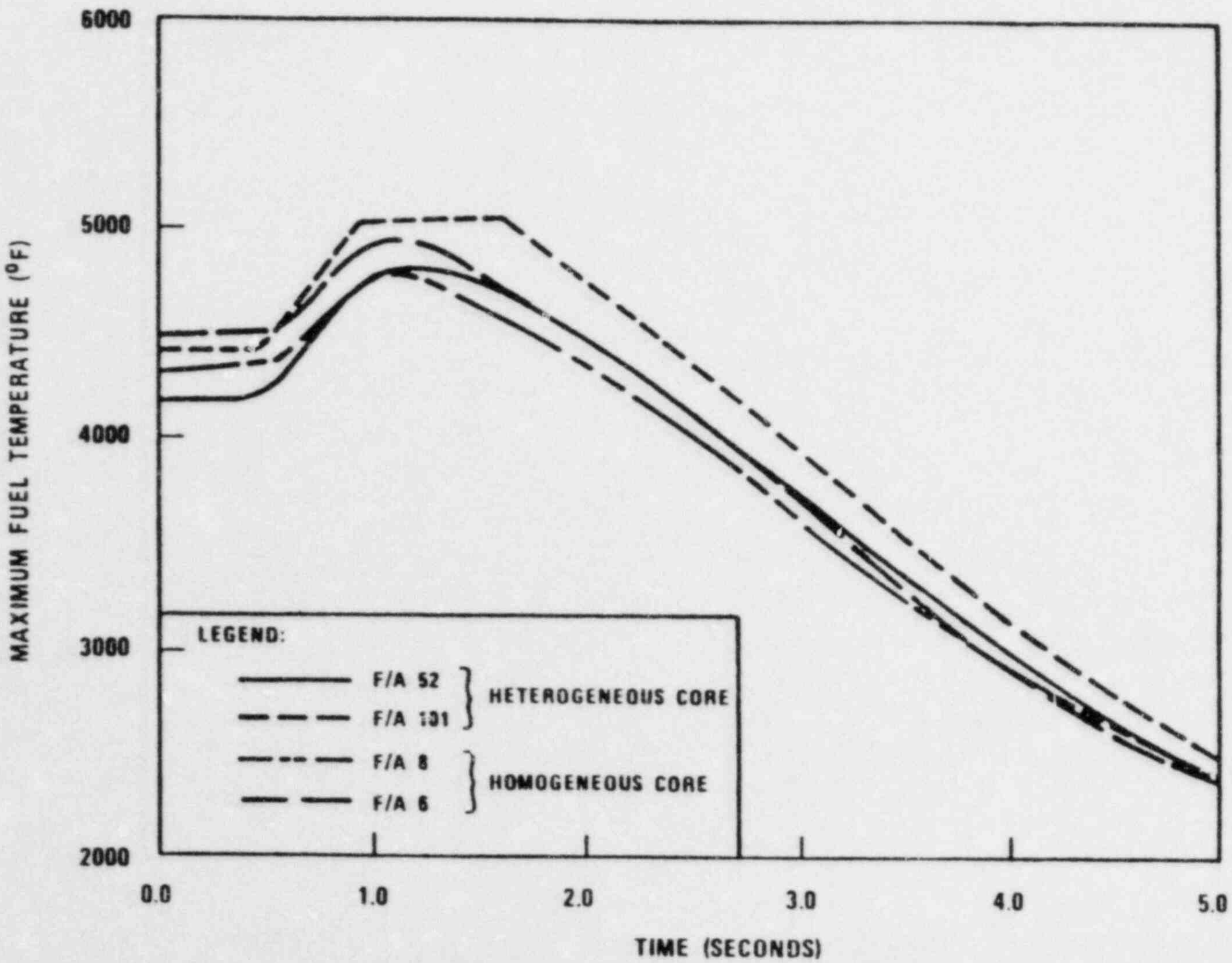


Figure 15.1.4-1 Variation of F/A Hot Channel Maximum Fuel Temperature for a 60¢ SSE Transient

1764-62

TABLE 15.6.1.4-3

SUMMARY RESULTS FOR SODIUM LEAKS INTO THE PHTS CELLS AND THE REACTOR CAVITY

Peak Transient Values

	Gas Pressure psig	Gas Temperature °F	Floor Structural Concrete Temperature °F	Wetted Wall Structural Concrete Temperature °F
PHTS Cells	14.4	680	190	200
Reactor Cavity	10.3	650	200	210

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Jan. 1982

TABLE 15.6.1.4-4

DESIGN BASIS RADIOACTIVE CONTENT
OF PRIMARY SODIUM COOLANT
30 YEARS REACTOR OPERATION

	μCi/gm Sodium Days After Shutdown			μCi/gm Sodium Days After Shutdown	
ISOTOPE	0	10	ISOTOPE	0	10
Na 24	2.94E+4*	4.32E-1	Te 127	2.48E-1	1.95E-1
Na 22	3.49E+0	3.46E+0	Te 127m	2.08E-1	1.95E-1
Rb 86	2.00E+0	1.38E+0	La 140	6.54E-2	3.80E-2
Cs 137	8.42E+1	8.42E+1	Ce 141	7.75E-2	6.26E-2
Cs 136	1.74E+1	1.05E+1	Ce 144	4.59E-2	4.48E-2
Cs 134	1.07E+1	1.06E+1	Pr 144	4.59E-2	4.48E-2
Sb 125	4.83E-1	4.80E-1	Pr 143	5.49E-2	3.30E-2
I 131	4.97E+1	2.10E+1	Nd 147	2.57E-2	1.38E-2
Te 132	3.53E+0	4.16E-1	Pm 147	2.57E-2	2.55E-2
I 132	3.35E+1	3.95E+0	Pu 238	1.60E-2	1.60E-2
Te 129m	7.18E-1	5.86E-1	Pu 239	4.24E-3	4.24E-3
Te 129	7.18E-1	5.86E-1	Pu 240	5.54E-3	5.54E-3
Sr 89	1.10E-1	9.60E-2	Pu 241	4.60E-1	4.59E-1
Sr 90	6.80E-2	6.80E-2	Pu 242	1.18E-5	1.18E-5
Y 90	6.80E-2	6.80E-2	Np 238	4.91E-6	1.80E-7
Y 91	3.13E-2	2.78E-2	Np 239	1.58E-2	8.23E-4
Zr 95	5.83E-2	5.24E-2	Am 241	1.64E-3	1.64E-3
Nb 95	5.83E-2	5.24E-2	Am 242m	6.46E-5	6.46E-5
Ru 103	8.31E-2	6.98E-2	Am 242	7.39E-5	6.46E-5
Ru 106	5.75E-2	5.64E-2	Am 243	2.64E-5	2.64E-5
Rh 106	5.75E-2	5.64E-2	Cm 242	1.20E-3	1.15E-3
Sb 127	3.65E+0	5.85E-1	Cm 243	1.59E-5	1.59E-5
Ba 140	6.54E-2	3.80E-2	Cm 244	3.32E-4	3.32E-4
			H3	2.34E+0	2.34E+0

*2.94E+4 = 2.94×10^4

TABLE 15.7-1

OTHER EVENTS

Section No.	Events	Potential Limiting Parameters	Comments
15.7	Other Events		
15.7.1	Anticipated Events		
15.7.1.1	Loss of One D C System	None	No adverse operating conditions have been identified with this event.
15.7.1.2	Loss of instrument or valve air system	None	Detailed description of failure effects or safety-related instrument air supplies, if any, will be provided in the FSAR.
15.7.1.3	IHX Leak	None	Core sees normal shutdown.
15.7.1.4	Off-normal cover gas pressure in the reactor primary coolant boundary	None	No adverse operating conditions associated with this event.
15.7.1.5	Off-normal cover gas pressure in IHTS	None	No adverse operating conditions associated with this event.
15.7.2	Unlikely events		
15.7.2.1	Inadvertent release of oil through the pump seal (PHTS)	None	No adverse consequence identified at this time.
15.7.2.2	Inadvertent release of oil through the pump seal (IHTS)	None	No adverse consequence identified at this time.
15.7.2.3	Generator breaker failure to open at turbine trip	None	Core sees only normal shutdown.
15.7.2.4	Rupture of RAPS Cryostill	<3 REM (integrated 2-hr dose at the site boundary)	Consequences would be within the suggested guideline doses.
15.7.2.5	Liquid rad-waste system failure	3.7×10^{-6} REM @ site boundary, 3.05×10^{-7} REM @ LPZ	Consequences would be within the suggested guideline doses.
15.7.2.6	Failure in the EVST NaK System	None	No adverse consequences associated with these events.
15.7.2.7	Leakage from sodium cold traps	7.8×10^{-5} REM @ site boundary, 2.3×10^{-5} REM @ LPZ	Consequences would be within the suggested guidelines doses.

TABLE 15.7-1

OTHER EVENTS (Cont'd.)

Section No.	Events	Potential Limiting Parameters	Comments
15.7.2.8	Rupture in RAPS Noble Gas Storage Vessel Cell	'<3 REM (Integrated 2-hr dose of the site boundary	Consequences would be within the suggested guideline doses.
15.7.2.9	Rupture in the CAPS cold box	0.14 REM @ site boundary	Consequences would be within the suggested guideline doses.
15.7.3	Extremely unlikely events		
15.7.3.1	Leak in a core component pot	~3200°F Center Fuel Pin	Only slight cladding melting. Fission gas release within umbrella of Section 15.5.2.3.
15.7.3.2	Spent fuel shipping cask dropped from maximum possible height	8.89×10^{-7} REM Whole Body @ SB (2-hr) 1.13×10^{-6} REM Whole Body @ LPZ (30-day)	Doses are well within the suggested guidelines.
15.7.3.3	Maximum possible conventional fires, flood, and storms	None	None
15.7.3.4	Failure of plug seals and annuli	None	No adverse consequences associated with this event.
15.7.3.5	.uel rod leakage combined with IHX and steam generator leakage	None	No adverse consequences associated with this event.
15.7.3.6	Sodium Interaction with Chilled Water	None	None
15.7.3.7	Sodium-Water reaction in large component cleaning vessel	0.01 REM @ site boundary	Consequences would be within the suggested guideline doses.

TABLE 15.7.2.4.-1

RUPTURE OF THE RAPS CRYOSTILL

Refueling Door Open - No Cell Leak Tightness Assumed

Isotope	Initial Inventory In the Cryostill (Ci)	Radioactivity Released From the Plant in 2 Hours (Ci)	0 to 2 Hours Whole Body Site Boundary Dose (Rem)
Xe133	4.67×10^5	3.92×10^4	1.38
Xe135	8.79×10^4	6.89×10^3	1.33
Kr88	1.66×10^3	1.11×10^2	0.169
Total	5.57×10^5	4.62×10^4	2.88

*There is an additional contribution of 0.09 rem from the daughter product of Kr88, which is Rb88.

TABLE 15.7.2.4-2

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Amend. 70
Aug. 1982

Chapter 17 - Quality Assurance

17.0 INTRODUCTION

17.0.1 Scope

This chapter describes the program of plans and actions to assure the quality of parts of the Clinch River Breeder Reactor Plant. These parts are those structures, systems, and components whose satisfactory performance is required to prevent accidents that cause undue risk to the health and safety of the public or to mitigate the consequences of such accidents if they were to occur. The program of plans and actions described herein is the Quality Assurance program for the Project. When the term "Program" is used hereafter in this chapter, it shall be understood to mean Quality Assurance Program unless otherwise defined at the point of use.

17.0.2 Quality Philosophy

Structures, systems and components of the plant referred to in Section 17.0.1 will be of the highest quality determined necessary consistent with their importance to plant safety. Quality as used here refers to those characteristics of items or services which collectively contribute to its ability to satisfy the requirements of its intended service. The quality of an item may be made up of many characteristics that can be identified and measured. For example, quality characteristics may be defined in terms of:

> Properties - Such as physical, dimensional, metallurgical, chemical, etc.
>
> State of Condition - Such as temperature, pressure, density, etc.
>
> Performance - Such as speed, duration, output, consumption, life expectancy, accuracy, efficiency, precision, etc.

and other similar attributes which are measurable quantities.

The ultimate quality of the plant will be the result of two basic functional processes. One which may be characterized as an "achieving" process and the other as an "assuring" process. The "achieving" functions are those work activities associated with planning, designing, manufacturing, constructing, operating, etc. The "assuring" functions are those activities associated with planning, controlling, inspecting, testing, surveillance, auditing, recording, etc. Within this combination of efforts, the overall quality of the plant is attained by all those work activities of an "achieving" nature with achievement assured through all those activities of an "assurance" nature. This latter function is the quality assurance function. Quality Assurance as used here refers to all those planned and systematic actions necessary to provide adequate confidence that an item will perform satisfactorily in service. In the broadest sense, both the "achieving" functions and the "assuring" functions are quality related activities.

Determining and specifying the quality requirements of the plant is an engineering function accomplished through planning and design. These requirements are defined through development of criteria, application of codes and standards, and the preparation of descriptions, drawings, specifications, procedures and instructions. The conversion of these plans and specifications into structures, systems, and components of the plant will be accomplished during manufacturing and construction. Much of the other chapters of this PSAR are concerned with these activities of the overall Project work program, and these activities will be carried out by the Project organizations as described in Section 1.4 of this PSAR. Any substantive changes in these organizational elements will be reported to NRC within 30 days after announcement.

The Quality Assurance Program for the Project is described in this chapter of the PSAR. Quality Assurance Program, as used here, refers to the overall integrated practice established and implemented to assure quality achievement. This overall program is described in Section 17.1.2. The major elements and activities of the program are illustrated in Figure 17.1-1. The delineation of responsibility and authority for this program is contained in Section 17.1.1 and illustrated in Figure 17.1-2.

17.0.3 Participants

The Project is a joint undertaking of the United States Government and the Electrical Utility Industry. Under the Project arrangements, major Project participants are as follows:

Breeder Reactor Corporation (BRC) - is responsible for (a) providing senior counsel to the Project, (b) arranging for widespread dissemination of information about the Project, and (c) obtaining and collecting the utility contributions and providing them to Project Management Corporation for use on the Project.

Project Management Corporation (PMC) - is responsible for administering the interest of the Utility Industry with respect to the Project. PMC is joined by the Tennessee Valley Authority and the United States Department of Energy for a joint licensing application, however, DOE is the lead agency for licensing purposes.

United States Department of Energy (DOE) - is responsible to contract, manage, and carry out the Project consistent with the principal objectives. This includes the design, construction, and demonstration of the plant. DOE will be the plant owner and lead applicant for licensing purposes. DOE will also supply the nuclear fuel for the plant.

Tennessee Valley Authority (TVA) - is responsible for plant operation and maintenance. TVA will join with PMC and DOE as an applicant for licensing purposes and may eventually take possession of the plant and its facilities.

Westinghouse Electric Corporation Advanced Reactors Division (ARD) - is the Nuclear Steam Supply System Supplier (NSSS/S) and is responsible for the overall design and manufacture of the Nuclear Steam Supply System (NSSS). ARD is also specifically responsible for the design and manufacture of reactor and reactor enclosure systems, primary sodium heat transport system, and related components and controls.

General Electric Corporation - Advanced Reactor Systems Department (GE-ARSD) - is a Reactor Manufacturer (RM) and is a major contractor for the Nuclear Steam Supplier, and as such is responsible for the design and manufacture of the intermediate heat transport system, and related systems and controls.

Rockwell International Energy Systems Group Atomics International Division (ESG-AI) - is a Reactor Manufacturer (RM) and is a major contractor for the Nuclear Steam Supplier, and as such is responsible for the design and manufacture of fuel handling systems, auxiliary sodium systems, reactor plant maintenance, and related systems and controls.

Burns and Roe Corporation (B&R) - is the Architect-Engineer (AE) for the overall plant including BOP and portions of the NSSS.

Stone and Webster Engineering Corporation (S&W) - is the Plant Constructor and will function as both a construction manager and a construction contractor including Procurement of Construction materials and services and selected BOP equipment.

17.0.4 Project Phase Approach

The Project Quality Assurance Program is planned and executed to match the phases through which Project work activities will progress. These are design, procurement, manufacturing, construction and operation.

For SAR purposes, the Quality Assurance Program is described in two parts. The first part includes those program practices that will be executed during design and construction where construction includes procurement, manufacturing and installation. This part is described in this PSAR. The second part will cover plant operation including preoperational testing, start-up and normal operations which will not be described until submittal of the FSAR, however, this part will be executed under a Quality Assurance Program which complies with Appendix B of 10 CFR Part 50.

The Owner program has been described to cover all Project phases and therefore includes practices applicable to design, construction, testing and operation.

17.0.5 Applicability

The described Quality Assurance Program contained in this chapter, including its Appendices, is intended for application to those safety-related structures, systems and components described in Sections 3.2, 7.1 and 9.13 and also to the reactor core and reactor vessel internals. The wording of the descriptions of the major participant programs in the Appendices to this chapter have not been restricted to this application alone. Thus, the practices described may be broader in scope and may be applied to areas of the plant other than those identified in Sections 3.2, 7.1 and 9.13 or to the reactor core and the reactor internals.

The Quality Assurance Program description contained in this chapter including its Appendices will as a minimum be reviewed and updated annually as appropriate and resulting modifications and updates will be reported to NRC. In addition, any changes to the Quality Assurance Program affecting this program description which changes or affects the authority, independence, or management reporting levels previously established for persons or units of the organization performing quality assurance functions will be reported to the NRC for review and acceptance prior to implementation. Other changes in organization elements will be reported to NRC within 30 days of such a change.

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTIOI.

17.1.1 Organization

17.1.1.1 Organization of Participants

The Owner role in the Project is the responsibility of DOE. To execute their responsibilities, the CRBRP Project Office has been established with staff provided from both PMC and DOE. Both PMC and DOE participate in CRBRP Project Office activities with respect to their designated responsibilities. Through this effort, the Owner role is fulfilled.

The CRBRP Project Quality Assurance Program is being or will be planned and implemented through the efforts of the major Project participants. The program will be established and conducted as described in Section 17.1.2. While development and execution of selected major portions of the program have been delegated to major Project participants, the effectiveness and adequacy of the program are the ultimate responsibility of the Owner.

To fulfill overall program responsiblity for assuring Quality achievement, there are certain quality assurance functions the Owner will perform. These are outlined in the Owner's program described in Appendix A of this section. The Owner's organization, established to perform these quality assurance functions, is also descr there.

For quality assurance program purposes (i.e., the "assurance" function), the quality assurance organizations of the major project participants are responsible to the Owner's Quality Assurance organization. This arrangement of major program participants is illustrated in Figure 17.1-2.

17.1.1.2 Responsibility and Authority

Each major participant has assigned the overall management responsibility for developing and executing its portion of the overall Quality Assurance Program to one member of management. In each major participant's organization, that individual responsible for the quality assurance function (the Quality Assurance Manager) resports directly to the Agency Official or Corporate Operations Officer who has overall responsibility for the Project (i.e., design, cost, schedule, administration, quality assurance, etc.)

The assignment of responsibility for program activities within each participant's organization is defined using organization charts, narratives, etc., with duties, responsibilities and authority of key personnel described and illustrated for each organizational component. These are provided in each major participant's program description contained in appropriate appendices of this chapter.

17.1.1.3 Communications

The free, continuous, unimpeded flow of communications between major participants both horizontally as well as vertically between all levels is essential. The free exchange of information between responsible individuals is essential to the expedient execution of quality assurance activities.

To promote the flow of communications and to assure the positive attention to quality problems, lines of communication are established between major quality assurance program participants in accordance with organizational arrangements shown in Figure 17.1-2.

17.1.1.4 Interface Control

A program-wide management review practice has been established to fulfill an interface coordination function. These reviews are executed at quality assurance program management review meetings. They provide a means for assessing project quality accomplishments, discussing program audits and resolving management problems.

These meetings are held quarterly with a schedule published on a yearly basis. The Owner representative chairs each meeting and such meetings are attended by management representatives of each major program participant. Any program participant may request a review meeting for any special purpose or problem should the need arise.

17.1.2 Quality Assurance Program

17.1.2.1 Program Requirements

The Quality Assurance Program for the Project complies with the requirements contained in RDT Standard F 2-2, "Quality Assurance Program Requirements" and supporting standards RDT F 1-2, "Preparation of System Design Descriptions", RDT F 1-3, "Preparation of Unusual Occurrence Reports", and RDT F 3-2, "Calibration Program Requirements". The program is in accordance with the requirements of Contract AT(49-1?)-12 between the United States of America as represented by the United States Department of Energy and the Tennessee Valley Authority, Commonwealth Edison Company and Project Management Corporation.

The individual in each major participant organization who is authorized to control further processing, delivery or installation of a nonconforming item, deficiency or unsatisfactory condition until proper disposition has occurred or to stop unsatisfactory work is required to be identified and the scope of his responsibility is required to be documented in writing.

The Project requirements for calibration records for measuring and test equipment are contained in RDT F 3-2 and are judged to include appropriate techniques for a program to assure the accuracy of measuring and test equipment and to provide confidence in the acceptability of items with characteristics determined and verified by controlled measuring and test equipment. Project participant procedures are required to reflect those requirements and are reviewed to determine that necessary activities are performed and documented such that the ultimate objectives are attainable. Participant implementation of procedures is monitored and evaluated to verify adequacy of performance. The specific information to be contained in the records of the activity include the data necessary to provide traceability and to permit corrective action if such were to become necessary. the Project's requirements and implementing practice provide methods for achieving that goal.

The Projects' requirements include general requirements for qualification of personnel such as those who perform inspection, examination and tests for acceptance purposes, including a requirement that records of personnel qualification status be maintained. While F 2-2 does not specifically require certification, it does not preclude the required records taking that form, therefore, persons who perform inspection, examination or test for acceptance purposes at the construction site shall be certified in accordance with the requirements of Section 2.2 of ANSI N45.2.6-1973.

In each major program participant's organization, personnel are selected and assigned their areas of responsibility based upon experience, education and management's assessment of their performance capabilities. They are observed for performance evaluation on a continuous basis by appropriate management. On-going training and indoctrination programs are conducted as described in the individual participants quality assurance program description.

ANSI N45.2.9 specifies requirements for collection, storage and maintenance of records. RDT F 2-2 specifies records requirements also, however, the records management system specified by F 2-2 does not include the same level of detail as the system specified by N45.2.9. In establishing the details of the Project's records management system necessary to implement the requirements of F 2-2, the Owner has directed that "the records management practice established as a part of each major participants program shall conform with ANSI N45.2.9."

RDT F 2-2 requires that qualification testing be directed "toward evaluation of the performance capability under various conditions as required by the design," and that it be conducted in accordance "with written and approved specifications and procedures." F 2-2 does not identify the requirement that qualification testing be performed to demonstrate adequacy of performance under the most adverse design conditions. However, it does require that specifications and procedures identify the detailed testing requirements including the most adverse conditions if they can be determined and are appropriate for the test objective. This also includes use of scaling laws and their verification when mock-ups are used in qualification testing.

The program described herein complies with the NRC-licensing requirements contained in Appendix B to 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The programs established by participants are structured to align with RDT Standard F 2-2, 1973. However, for ease of NRC-licensing review a matrix has been prepared (Figure 17.1-3) to show correlation of RDT F 2-2 requirements with the 18 criteria of Appendix B to 10 CFR 50. To further assist in licensing review, the major program participants descriptions have been aligned to correspond with the appropriate criteria of Appendix B to 10 CFR 50.

Although the various provisions of the regulatory guides listed in Subsection VI of the Standard Review Plan are not established as requirements for the CRBRP Quality Assurance Program, the Owner will accept practices that comply with these guides as fulfilling the like requirements of RDT Standard F 2-2 insofar as the requirements of RDT F 2-2 are met. Therefore, in implementing the program described herein, maximum recognition will be made of quality assurance practices in accordance with NRC Regulatory Guides and nationally recognized codes and standards such as ANSI N45.2, ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, and others.

17.1.2.2 Transfer of Requirements to Lower Tier Participants

Quality assurance program activities required of participants will be specified in contracts. These contracts will include the appropriate requirements of 10 CFR 50, Appendix B. These requirements may be specified directly in contract scope of work statements, referenced specifications or referenced code or standard. It is recognized that participants may have established quality assurance programs in accordance with Appendix B to 10 CFR 50, ANSI N45.2 or other nationally recognized codes and standards. These programs will be recognized to the maximum extent possible as acceptable methods of complying with the specified requirements.

17.1.2.3 Overall Quality Assurance Program Concept

To fulfill the requirements for quality assurance, the Project has established an overall integrated program of quality assurance plans and actions in which every participant has a part. The policies and objectives of the program have been defined as follows:

1. To assure the attainment of the level of quality necessary for the accomplishment of the Project objectives commensurate with the Owner's responsibility for protection of the public health and safety, and for the protection of the environment.

2. To assure that facilities, structures, systems, components and equipment; designed, procured, fabricated, installed, constructed, tested, operated, or modified by or for the Owner conform to specified requirements.

3. To assure that appropriate quality assurance activities are implemented by or for the Owner.

17.1.2.4 Organization of Quality Assurance Program Participants

To conduct the overall Quality Assurance Program, participation has been organized into three levels, as illustrated in Figure 17.1-4. By licensing requirements, the Owner is responsible for the overall program and its adequacy. The Owner portion of the program is at the third level and is primarily a management-type program with surveillance, interface coordination, and program integration functions. The second level of the overall program includes the other major program participants that have direct or indirect interface with the Owner. These are:

The Fuel Supplier	(FS)
The Nuclear Steam Supply System Supplier	(NSSS/S)
The Reactor Manufacturers	(RM)
The Architect-Engineer	(AE)
The Balance of Plant Supplier	(BOP Supply)
The Constructor	(CONSTRUCTOR)

These portions of the overall program are also management-type programs with surveillance, interface coordination, and lower tier program integration functions. The first level of the overall program includes a multitude of systems, component, material, and service suppliers. Their quality assurance programs are primarily work practice-oriented programs concerned with direct control and verification through inspection, examination and testing.

17.1.2.5 Organization of Quality Assurance Program Elements

The major elements of the overall Quality Assurance Program are identified in Figure 17.1-1. These have been grouped by project phase or function in which they occur or to which they relate. This figure makes no attempt to identify which program participant will perform which activity, but is intended to illustrate the total overall scope of program activities that will be performed during the life of the program. It should also be pointed out that these elements are programmatic or management systems type activities and should not be interpreted as containing all the detailed work practice oriented activities for performing such things as special process controls or specific methods of inspection, examination, or testing. For ease of communication and reference, each participant's portion of the overall program is referred to as that participant's quality assurance program. It is recognized that the scopes of these programs will vary widely. Each program was or will be developed and implemented consistent with the extent of participation. Each participant's program will include management practice as described in RDT F 2-2, Section 2 and 8, and other programmatic practices as appropriate to the scope of his participation.

Owner Program - Major elements of the Owner Quality Assurance Program are shown in Figure 17.1-5. These include the management practice and those unique activities of overall program responsibilities. This program is described in Appendix A to this chapter.

FS Program - Major elements of the FS Quality Assurance Program are shown in Figure 17.1-6. These include the program management practices complimented by those practices unique to procurement and manufacturing. This program is described in Appendix B to this chapter.

BOP Supply Program - Major elements of the BOP Supply Quality Assurance Program are shown in Figure 17.1-7. These include the management practice complemented by those practices unique to procurement and manufacturing. This program is described in Appendix C to this chapter.

NSSS Supplier/RM Programs - Major elements of the NSSS Supplier/RM Quality Assurance Programs are shown in Figure 17.1-8. These include the management practices complemented by those practices applicable to design, procurement, and manufacturing. These programs are described in Appendix D, NSSS Supplier; Appendix I, GE-RM; and Appendix J, AI/ESG RM.

AE Program - Major elements of the AE Quality Assurance Program are shown in Figure 17.1-9. These include the management practice complemented by those practices applicable to design, procurement and manufacturing. This program is described in Appendix E to this chapter.

CONSTRUCTOR Program - Major elements of the constructor Quality Assurance Program are shown in Figure 17.1-10. These include management practices complemented by those practices unique to procurement, manufacturing, and construction. This program is described in Appendix F to this chapter.

17.1.2.6 Scope of Quality Assurance Program Application

The overall Quality Assurance Program described in this section is, or will be applied to the planning, design, procurement, manufacturing and construction of the safety-related parts of the plant identified in Section 3.2, 7.1 and 9.13 and also to the Reactor Core and the Reactor Vessel Internals. The elements applied and the extent to which they apply will vary but all safety-related structures, systems and components listed will be covered.

17.1.2.7 Status of Project Work and Program Application

On July 25, 1973, agreements were signed by the principals in the Clinch River Breeder Reactor Plant (CRBRP) Project, the U. S. Atomic Energy Commission, Commonwealth Edison Company of Chicago, and the Tennessee Valley Authority officially forming the two, non-profit organizations created in 1972 to carry out the project: The Breeder Reactor Corporation, representing the sponsoring utilities; and the Project Management Corporation, representing the principal electric industry and government agency partners. On November 14, 1973, Project Management Corporation entered into the first of two prime contracts for demonstration plant design services with Westinghouse Electric Corporation which covers the design, manufacture, furnishing, and testing of the Nuclear Steam Supply System (NSSS) for the 350 to 400 megawatt breeder demonstration plant. Under the provisions of the contract, Westinghouse and its Advanced Reactors Division are being assisted by the General Electric Company and the Atomics International Division of Rockwell International as subcontractors.

The second prime contract for demonstration plant design services, with Burns and Roe, Inc., as architect-engineer was signed January 25, 1974. This contract covers management, engineering, design and drafting services for the overall plant and site layout, nuclear-island building arrangements and general plant systems and structures. Actual project work was started by Westinghouse and Burns and Roe in February 1973 under interim arrangements. In March 1973, Westinghouse Environmental Systems Department was assigned responsibility (with input from Project participants) for preparation of the CRBRP Environmental Report. In early 1973, site investigation activities commenced in order to gather data for the PSAR. In January, 1974, study contracts were initiated with Combustion Engineering and Foster Wheeler Corporation for major plant components. This same month, specifications were prepared for the turbine-generator and requests for proposals were sent to manufacturers. In March 1974, evaluations of the submitted turbine generator proposals were begun. Since then, a letter of intent has been signed with General Electric Company for this procurement. In June 1974, Westinghouse issued a report on the reference design for the plant. Preliminary site investigations were also completed during this month. Commencing in October

1974, purchase orders for long lead materials were issued for reactor vessel steel plate material, IHX tube sheet forgings and stainless steel plate material, stainless steel forging for the core support structure, forged flange and bar forgings for the IHX and stainless steel piping for the core support structure. RFP's for the steam generator plant units were issued and proposals were received from Atomics International and Foster Wheeler Corporation.

Also, on October 15, 1974, the CRBRP Construction Permit and Operating Licensing Application, supported by the Environmental Report, Chapter 2 of the Preliminary Safety Analysis Report, Statement of General Information and Description of Site Preparation Activities, was tendered with AEC Regulatory.

For quality assurance in particular, contract AT(49-18)-12 established the requirements for the quality assurance program as RDT Standard F 2-2, quality assurance program requirements. The program was established in 1973 with participation organized into a three level structure as shown in Figure 17.1-4.

In early 1974, the elements of the overall program that should be in place to cover early project design and procurement activities were identified. Major participants were required to have these practices in place by mid-May to cover the design and early procurement activities. Audits commenced in July and August to determine if the requisite quality assurance program has been implemented, and to insure the requirements of RDT F 2-2 are being met with respect to design control. Audit efforts were conducted on a continuing basis in accordance with specific quality assurance plans and where a need was determined by PMC and/or the NSSS-LRP (DOE).

17.1.3 References Referred to in the Text

* Standard RDT F 2-2, "Quality Assurance Program Rquirements", with Amendments I, December 1973, 2, March 1974, and 3, July 1975. (RDT F 2-2)

* 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

* NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 2, July 1981

* ASME Code, "ASME Boiler and Pressure Vessel Code," Section III, Nuclear Power Plant Components, 1974.

* ANSI N45.2 - 1971, "American National Standard, Quality Assurance Program Requirements for Nuclear Power Plants."

* Standard RDT F 1-2T, "Preparation of System Design Descriptions," March 1972, with Amendment 1 (RDT F 1-2T).

* Standard RDT F 1-3T, "Preparation of Unusual Occurrence Reports," February 1974, with Amendment I and 2 (RDT F 1-3T).

* Standard RDT F 3-2T, "Calibration Program Requirements," Feburary 1973 (RDT F 3-2T).

17.1.4 <u>Acronyms used in Chapter 17 Text and Appendices</u>

ADM	-	Administrative Procedures (WARD)
AD/PR	-	Assistant Director for Procurement
AE A/E	-	Architect Engineer
AEC	-	Atomic Energy Commission
AMEND	-	Amendment
ANSI	-	American National Standards Institute
ANS	-	American Nuclear Society
APP	-	Appendix
ARD	-	Westinghouse Advanced Reactors Division
ARSD	-	Advanced Reactor Systems Department (GE)
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing Materials
BOP	-	Balance of Plant
B&R	-	Burns and Roe, Inc.
BRC	-	Breeder Reactor Corporation
BRD	-	Breeder Reactor Division
CAR	-	Corrective Action Request
CECO	-	Commonwealth Edison Company
CFR	-	Code of Federal Regulations
CINDT	-	Controlled Information Data Transmittals
CRBRP	-	Clinch River Breeder Reactor Plant
D	-	Design & Drafting Procedures (B&R)
DCC	-	Document Control Center (GE)

DEC	-	December
DOC	-	Document
DOE	-	United States Department of Energy
E	-	Engineering
"E"	-	Equipment
ECN	-	Engineering Change Notice
ECP	-	Engineering Change Proposal
EDL	-	Engineering Drawing List
e.g.	-	For Example (exempli gratia)
EM	-	Environmental Monitor
EMP	-	Engineering Management Procedures (ESG)
ENG's	-	Engineering Procedures (GE)
ERM	-	Engineering Review Memorandum (GE)
ESG	-	Energy Systems Group (Rockwell International)
E-Specs	-	Equipment Specifications
etc.	-	and so forth (et cetera)
FS	-	Fuel Supplier (Richland Operations Office of DOE)
FSAR	-	Final Safety Analysis Report
F&TPs	-	Fabrication and Test Procedures (GE)
GE	-	General Electric
GE-ARSD	-	General Electric (Advanced Reactor Systems Department)
ICD	-	Interface Control Drawings
i.e.	-	that is - (id est)
IEEE	-	Institute of Electrical & Electronic Engineers
L	-	Licensing Procedures (B&R)
LRP	-	Lead Role Participant
LX	-	Project Internal Procedures (WARD)

M&MM	-	Manufacturing Manual Procedures
MDMs	-	Material Division (Purchasing) Manual Procedures (ESG)
MPIs	-	Manufacturing Process Instructions (GE)
MPO	-	Manufacturing Production Orders
MPS	-	Management Procedures System
MR	-	Material Request
NDE	-	Nondestructive Examination
NIR	-	Nonconforming Item Record (GE)
No.	-	Number
NRB	-	Nonconformance Review Board
NSSS	-	Nuclear Steam Supply System
NSSS/S	-	Nuclear Steam Supply System Supplier (Westinghouse-Oak Ridge)
NSSS	-	Nuclear Steam Supply System Supplier Supplier/RM Reactor Manufacture Relationship
OPDD-10	-	Overall Plant Design Description
OPR	-	Operating Procedures (WARD)
PC	-	Project Control Procedures (B&R)
PMC	-	Project Management Corporation
PMD	-	CRBRP Project Management Directives (ESG)
POCN	-	Purchase Order Change Notice
PPL's	-	Program Planning Instructions (GE)
PRCN	-	Purchase Requisition Change Notice
PSAR	-	Preliminary Safety Analysis Report
PSC	-	Project Steering Committee
PUR	-	Purchasing Procedures
QA	-	Quality Assurance
QAOP	-	Quality Assurance Department Operating Procedures (ESG)

QAP's	-	Quality Assurance Acceptance Procedures (ESG)
QAPI	-	Quality Assurance Program & Procedures Index (GE)
QAPP	-	ASME Code Section III Quality Assurance Procedure (ESG)
QA/QC	-	Quality Assurance Quality Control
QCI	-	Quality Control Instruction (GE)
QCPP	-	ASME Code Section VIII Quality Assurance Manual (ESG)
QMP's	-	Quality Methods and Procedures (WARD)
QSI's	-	Quality Standing Instructions (GE)
RDT	-	Reactor Development & Technology (Former Division of Atomic Energy Commission superseded by RRD of USERDA - Standards Still Maintain RDT Title)
REF	-	Reference
REV	-	Revision
RFP	-	Request for Proposal
RFQ	-	Request for Quotation
RIP	-	Receiving Inspection Plan (GE)
RL	-	Richland Operations Office of DOE
RM	-	Reactor Manufacturer (ARD, GE, ESG-AI)
RRD of USERDA	-	Reactor Research & Development Division of United States Energy Research and Development Administration
SAR	-	Safety Analysis Report
SDD	-	System Design Description
SNT	-	American Society for Nondestructive Testing
SNT-TC	-	American Society for Nondestructive Testing 1A Recommended Practice
SOP's	-	Standard Operating Policies (ESG)
STPI's	-	Sodium Technology Process Instructions (GE)
TVA	-	Tennessee Valley Authority
UCR	-	Unsatisfactory Condition Record (GE)

U.S. - United States

VCR - Vendor Case Record

WARD - Westinghouse Electric Corporation Advanced Reactors Division

WASH - Washington

THE CLINCH RIVER BREEDER REACTOR PLANT

CHAPTER 17.0 - QUALITY ASSURANCE

APPENDIX A

A DESCRIPTION OF THE OWNER

QUALITY ASSURANCE PROGRAM

CLINCH RIVER BREEDER REACTOR PLANT
A DESCRIPTION OF THE OWNER
QUALITY ASSURANCE PROGRAM

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CLINCH RIVER BREEDER REACTOR PLANT
A DESCRIPTION OF THE OWNER
QUALITY ASSURANCE PROGRAM

0. INTRODUCTION

0.1 SCOPE

Contained herein is a description of the plans and actions by the Owner to assure the quality of structures, systems, and components of the Clinch River Breeder Reactor Plant (CRBRP). These plans and actions constitute the Owner's Quality Assurance Program.

0.2 BASIS

The program described herein has been planned, structured and defined to fulfill the responsibility for ultimate effectiveness and adequacy of the overall Project Quality Assurance Program. Responsibility for establishing and executing portions of the overall Project Quality Assurance Program has been or will be delegated to others participating in the Project with ultimate responsibility for the adequacy of their performance retained by the Owner.

0.3 APPLICATION

The Owner's Quality Assurance Program described herein is applicable to the planning, design, procurement, manufacturing, construction including testing, and operation of those safety-related structures, systems, and components identified in Section 3.2, 7.1 and 9.13 of this PSAR. For the Owner's purposes the program described herein is not limited to this application alone, however, and is or will be appropriately applied by the Owner to the CRBRP in its entirety including all structures, systems and components where satisfactory performance is required for the plant to operate reliably, safely and with minimum environmental effects.

1.0 ORGANIZATION

The Owner is RESPONSIBLE for the overall management of the Project to design, build, and operate the Clinch River Breeder Reactor Plant (CRBRP). The execution of this responsibility rests with the CRBRP Project Director, who is the principal operations officer of the Owner. Part of this responsibility is to assure that the plant is designed, built and operated in a way that will provide adequate confidence that it will perform satisfactorily in service. To provide for this assurance, the Project Director has directed the establishment and conduct of an overall integrated quality assurance program which shall have the objectives, carry out the functions, and be executed as hereafter defined.

1.1 FUNCTION

The functions that the Owner will perform in order to achieve the stated objectives of the Quality Assurance Program and fulfill its ultimate responsibility for program adequacy are as follows:

1. Development of an overall plan for conduct of the quality assurance program.

2. Assignment of program execution responsibility to appropriate program participants. These include the contractors and subcontractors who participate in the Project, as well as service contractors who only perform quality assurance activities.

3. Development of working plans and procedures to conduct program activities.

4. Organizing and staffing appropriately to implement program functions.

5. Implementation of Owner program activities.

6. Interfacing of major participant programs with the Owner's program.

7. Integration, coordination, evaluation, and approval of major participant quality assurance programs.

8. Development and implementation of major participant programs where the Owner elects to retain execution responsibility in lieu of assigning it to another organization, e.g., Balance of Plant Supplier.

Those quality assurance functions which have been delegated to other organizations are explained in Section 2.3 of this program description.

1.2 RESPONSIBILITY AND AUTHORITY

The Chief, Quality Assurance, serving as head of the CRBRP Project Office Quality Assurance Division and reporting directly to the Project Director, is assigned responsibility for devising, recommending, establishment of, and assuring effective execution of the overall Project Quality Assurance Program. The Chief, Quality Assurance, is responsible that organizations, systems, and procedures at all levels will provide assurance that the Demonstration Plant is designed in accordance with requirements, is constructed as designed, and is operated in accordance with plans and procedures to achieve the demonstration objectives. In carrying out these responsibilities, he is authorized by the Project Director to:

1. Identify quality problems.

2. Initiate, recommend, or provide solutions through designated channels.

3. Verify implementation of solutions.

4. Determine the adequacy of facilities and equipment provided to carry out approved procedures and instructions.

5. Authorize issuance of special instructions necessary to execute his responsibilities.

6. Notify responsible management of unsatisfactory work or unapproved practices and if necessary, stop unsatisfactory work or control further processing, delivery, or installation of nonconforming materials.

The Chief, Quality Assurance, is responsible for organizing the overall Project Quality Assurance Program and for recommending further assignments of execution responsibility as appropriate. He shall secure Charter statements from other major program participants describing their responsibilities and functions. He shall assure that in each major participant's organization the person responsible for quality assurance is granted sufficient authority to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions.

The Chief, Quality Assurance, is responsible for recommending to the Director the organization and staffing plan for the Quality Assurance Division in the conduct of quality assurance practices necessary to fulfill the Owner responsibilities for establishment and adequacy of the program. In this position, he is responsible for the technical and administrative control (except for DOE personnel) of individuals and groups within the Quality Assurance Division performing quality assurance activities or verifying adequacy in the performance of quality assurance related activities of others. The Project Office retains administrative control of DOE personnel.

1.3 ORGANIZATIONAL ARRANGEMENTS

The Owner's organizational structure for performing quality-related activities associated with management, planning, design, procurement, construction and operation of the CRBRP and the responsibility and authority of key positions within the organization are described in Section 1.4 of the PSAR. The Owner organization is shown in Figure 1.4-1.

To perform the assigned quality assurance functions, the Quality Assurance Division is organized as shown in Figure 17A-1. The Division is subdivided along functional lines to perform quality verification, quality engineering and quality improvement. A description of the organizational elements are contained in subsequent paragraphs.

The personnel of the division are located in Oak Ridge, Tennessee, in the CRBRP Project Office, located at Jefferson Circle. All Owner program activities will be executed by personnel working out of that office. Verification that activities by contractors are in compliance with requirements will be performed through surveillance, inspection and audit practices by personnel working out of the Project Office.

The staffing plan for the division is based on the Project's work plan and schedule. It reflects the planned scope of work to be performed and the quality assuring activities to be applied to that work. The number of people and their required capabilities are identified through this planning, scheduling, and resource estimating process.

1.3.1 Quality Verification Branch

The function of the Quality Verification Branch is to maintain surveillance over quality assurance programs of major Project participants and to verify quality achievement in their work performance. This branch is also responsible for monitoring the Owner's Quality Assurance Program to verify overall adequacy.

The Quality Verification Branch performs three types of activities as described below:

1.3.1.1 Surveillance

Monitoring of the Project work and the quality assurance practices on that work is performed through this activity. It also serves as the focal point for interface coordination between the Owner's Quality Assurance Program and the quality assurance programs of other Project participants.

1.3.1.2 Inspection

Inspection of items and services or the monitoring of inspections by others is accomplished through this activity. This activity includes performance of selected civil, structural, electrical, mechanical and welding inspections, and nondestructive examinations.

1.3.1.3 Audit

Planning and conducting internal audits of the Owner's Quality Assurance Program and external audits of contractor Quality Assurance Programs is accomplished through this activity. Scheduled and unscheduled audits are conducted.

1.3.2 Quality Engineering Branch

The function of the Quality Engineering Branch is to plan, define, and develop the overall Project Quality Assurance Program and the Owner's portion of that program. This function includes the preparation and maintenance of overall Project Quality Assurance Program requirements and internal plans and procedures. This branch has lead responsibility for quality assurance program progress and status reporting and quality records management.

The Quality Engineering Branch performs three types of activities as described below:

1.3.2.1 Planning

Planning, program development and the documentation of plans and procedures for conduct of both the overall Project Quality Assurance Program and the Owner's Quality Assurance Program are performed through this activity. A knowledge of industry and government standards and their appropriate application to the Project Quality Assurance Program is maintained.

1.3.2.2 Reports

Establishment of quality assurance program progress and status reporting requirements and their maintenance is accomplished through this activity. Collection of reports from branches of the Quality Assurance Division and the preparation of the Owner Quality Assurance Program Progress and Status Report is also performed.

1.3.2.3 Records

Collection, filing, and maintenance of quality records is performed through this activity. The receiving, routing, and filing of working documentation within the Quality Assurance Division is performed. The Quality Records File will ultimately include records of the overall CRBRP Project.

1.3.3 Quality Improvement Branch

The function of the Quality Improvement Branch is to provide needed training and indoctrination for Quality Assurance Division Personnel and to coordinate Project Office and Project-wide training and indoctrination activities for personnel performing quality-related functions. This branch is also responsible for conducting activities wherein nonconformances are dispositioned and corrections to program deficiencies are made to improve quality achievements and to prevent recurrence of nonconforming conditions.

The Quality Improvement Branch performs three types of activities as described below:

1.3.3.1 Nonconformance Control

Collecting unusual or abnormal occurrence reports, deviation requests, nonconformance reports, and deficiency citations, and processing them to satisfactory resolution is accomplished through this activity. A log of quality problems identified internally and by major program participants will be maintained and corrective actions recorded.

1.3.3.2 Trend Analysis

Activities, reports (audit, inspection, progress, status) and records are monitored through this activity to identify quality problems. Problems identified are studied and actions recommended to correct the problem, to improve quality achievements, and to improve the efficiency and effectiveness of quality assurance activities.

1.3.3.3 Training and Indoctrination

Actions to acquaint Project personnel with the various elements of the quality assurance program and the practices needed to assure quality achievement are performed through this activity. The certification of quality assurance personnel qualifications is performed and personnel training and indoctrination within the Project Office and within other Project participant organizations is monitored to assure that:

1. Personnel performing activities affecting quality are appropriately trained in the principles, techniques and requirements of the activity being performed.

2. Personnel performing activities affecting quality are instructed as to purpose, scope, and implementation of governing manuals, policies, and procedures.

3. Appropriate training procedures are established.

4. Indoctrination and training activities are conducted in an effective manner and achieve desired results.

5. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

6. Proficiency evaluations or tests, as appropriate, are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

7. Certificate of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria used to qualify personnel in each function.

8. Proficiency of personnel performing and verifying activities affecting quality is maintained through work experience or retraining with continued proficiency verified through reevaluating, reexaming, and/or recertifying in accordance with Project requirements.

1.4 QUALIFICATION REQUIREMENTS FOR QUALITY ASSURANCE MANAGEMENT POSITIONS

1.4.1 Chief, Quality Assurance

The individual assigned to retain overall authority and responsibility for the Owner's Quality Assurance Program is the Chief, Quality Assurance, who is the functional manager for directing and managing the Quality Assurance Program. He will have the following qualifications:

Education - He shall be a graduate of a four-year accredited engineering or science college or university.

Experience -

General - He shall have a minimum of 10 years experience in quality assurance or engineering, construction, or operation activities associated with nuclear facilities or equivalent heavy industry. A minimum of six years experience shall be in quality assurance.

Specialty - He shall possess a broad knowledge and understanding of industry and government codes, standards, and regulations defining quality assurance requirements and practices.

He shall have a broad knowledge and understanding of quality assurance methods and their application.

He shall have experience in planning, defining and performing quality assurance practices and the application of procedures.

Managerial - He shall be experienced in organizing, directing and administering an overall program of activity or a major portion of an overall program having broad scope and application.

He shall have experience in the supervision of personnel and the planning and management of other resources normally needed to conduct an extensive quality assurance program.

1.4.2 Chief, Quality Verification, Quality Engineering, and Quality Improvement

The individuals assigned to manage Quality Verification, Quality Engineering and Quality Improvement activities will have the following qualifications:

Education - He shall be a graduate of a four-year accredited science or engineering college or university.

Experience -

General - He shall have a minimum of five years experience in quality assurance or engineering, construction, or operation activities associated with nuclear facilities or equivalent heavy industry.

Specialty - He shall have a broad understanding and knowledge of applicable industry and government codes, standards and regulations defining quality assurance requirements and practices. He shall have experience in planning, defining, and performing quality assurance practices and the application of procedures to the area of work in which he is responsible.

Managerial - He shall be experienced in organizing, directing, and administering an overall program of activity or a major portion of an overall program having broad scope and application. He shall have experience in the supervision of personnel and be capable of directing and coordinating the activities of the contractors to achieve objectives.

1.5 COMMUNICATIONS

The free, continuous, unimpeded flow of communications, both horizontally and vertically within organizations as well as between the Owner and other program participants, is essential. The free exchange of information between responsible individuals is also essential to the expedient execution of quality assurance activities.

To promote the flow of communications and to assure positive attention to quality problems within the Project's Quality Assurance Program, lines of communication between the Owner and the organization of the other major program participants are established as follows:

1. Communications by Senior Management - These communications will deal with such matters as major changes in the scope of the Quality Assurance Program. Communications will be addressed to the responsible senior management official with copies to the major program participants cognizant of the subject.

2. Communications by Quality Assurance Program Management - These communications will provide a direct formal or informal exchange of information between the Owner's Quality Assurance Chief and other quality assurance managers of organizations which have a direct interface with the Owner. Copies of such communications will be distributed to other major program participants cognizant of the subject.

3. Communications by Individuals - These communications are encouraged to identify and evaluate quality problems and to initiate, recommend, or provide solutions. Communications may also be formal or informal, the choice of which shall depend on the significance of the subject and the judgement of the individuals involved.

Communication of quality assurance related activities within the Owner's organization is promoted through:

1. Periodic staff meetings of the Project Director.

2. Monthly quality assurance program progress and status reporting.

Communication of quality assurance related activities between the Owner's organization and the organizations of the major Program participants is promoted through:

1. Quarterly management review meetings attended by the Quality Assurance Managers of the major program participants.

2. Monthly quality assurance program progress and status reporting by major program participants.

2.0 QUALITY ASSURANCE PROGRAM

The Owner has established and is conducting a quality assurance program that meets the criteria of 10 CFR 50, Appendix B. A description of the activities of this program is contained herein and is presented to demonstrate how the Owner is meeting the applicable criteria.

2.1 POLICY AND OBJECTIVES

The policy and objectives of the Project Quality Assurance Program are:

1. To assure the attainment of the level of quality necessary for the accomplishment of the Project objectives commensurate with Owner responsibility for protection of the public health and safety, for the protection of the environment, and for reliable plant operation.

2. To assure that facilities, systems, components, and equipment designed, procured, fabricated, installed, constructed, tested, operated or modified by or for the Owner, conform to specified requirements.

3. To assure that appropriate quality assurance activities are implemented by or for the Owner.

2.2 RESPONSIBILITY

2.2.1 Owner

As described in Section 1.0 of this appendix, the Owner's principal operations officer for the Project is the Project Director. The Director has day to day management overview involvement in the Project Quality Assurance Program and the execution of the Owner's portion of the program.

Responsibility for the execution of the Owner program rests with the CRBRP Project Office organizational elements as detailed in the Owner's implementing procedures. The quality assurance program management practices are performed primarily by the Quality Assurance Division, while other Divisions of the CRBRP Project Office are primarily responsible for performing selected technically-related functions that are required for proper execution of program activities.

The technically-related functions that are performed by other Divisions of the CRBRP Project Office include:

- o Participation in multi-discipline audits and reviews.
- o Review and acceptance of program plans and procedures.
- o Review and concurrence with proposed corrective actions resulting from audit findings of deficiency.
- o Execution of certain design control and document control activities.
- o Execution of certain contracting control activities.

To determine program status and to evaluate program adequacy, the Owner executes an overall program review practice to assess the scope, implementation and effectiveness of the Quality Assurance Program. This review is performed by the Owner quality assurance organization at least annually using resources and information at its disposal, including the results from surveillance, inspection and audit activities.

The Project Director receives the results of these management reviews in the form of regular progress and status reports and other special reports as appropriate. These reports outline the progress and status of quality assurance activities, problems and nonconformances, quality trends and results of audits. The Project Director reviews these reports and initiates whatever management action is required to improve conditions and further implement the program.

In addition to the Project Director's review and assessment of the quality assurance program, an annual review and evaluation of the Project including the quality assurance program, is performed by a select committee appointed by the Breeder Reactor Corporation (BRC). The program is also subject to periodic review by the Project Steering Committee (PSC), see Section 1.4 of the PSAR. This review may be performed by the PSC itself, or by some other organization on an ad hoc basis as they may choose.

2.2.2 Participants

Although the Owner retains the responsibility for adequacy of the entire Project Quality Assurance Program, other major participants are assigned by contract the responsibility for establishing and implementing particular program practices. These delegated functions are described in Section 2.3.3.

Program participation responsibilities are organized in a three level structure illustrated in Figure 17A-2. The Owner portion of the program is at the third level. The other major program participants shown at the second level, and having a direct or indirect interface with the Owner are:

The NSSS Supplier	(NSSS/S)
The Reactor Manufacturer	(RM)
The Architect-Engineer	(AE)
The Fuel Supplier	(FS)
The Balance of Plant Supplier	(BOP Supply)
The Constructor	(CONSTRUCTOR)
The Operator	(OPERATOR)
The Environmental Monitor	(EM)

At the first level are the multitude of systems, components, materials, and service suppliers.

2.3 REQUIREMENTS

2.3.1 Overall

The requirements for the overall Quality Assurance Program are contained in RDT Standard F 2-2, 1973, with Amendments 1, 2, and 3, Quality Assurance Program Requirements, and are illustrated by the identification of the program elements shown in Figure 17A-3. Execution of a program which meets these requirements will comply with 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

2.3.2 Program Elements Executed by the Owner

The elements of the overall Quality Assurance Program which will be executed by the Owner are those identified as "Overall Program" and "Program Management" shown in Figure 17A-4. This is the Owner portion of the program, and it will be executed in accordance with plans, procedures and management practices of the Owner. Figures 17A-12 and 17A-13 contain an index of the Owner plans and procedures and indicates the program elements as defined in RDT F 2-2 and 10 CFR 50, Appendix B respectively that will be implemented in accordance with specific documents. A listing of these procedures with a brief description of each is contained in Attachment 1. These documents are contained in the Owner's Manuals which have been developed to implement the quality assurance policies, goals and objectives of the Owner Quality Assurance Program.

2.3.3 Program Elements Delegated to Others

The responsibility for execution of various other program activities is delegated by the Owner to other major program participants and are applied to their respective scopes of participation as appropriate. Those major participants may pass along those activities to subcontractors so that each activity can be performed by those most qualified. The ultimate responsibility for adequate implementation and performance by each participant is retained by the Owner. The Owner requires that each major participant document his program in appropriate descriptions, plans and procedures. Each program is initially evaluated and approved by the Owner. Subsequently, each participant's revised, updated or new program plans and procedures are reviewed on an ongoing basis and are accepted for Project use contingent upon implementation resulting in conformance to Project requirements. Each major participant's program is monitored on a continuing basis through review and audit to assess its adequacy and to verify compliance with Project requirements including provisions for special equipment, environmental conditions, skills, or processes as may be necessary.

The major elements of the FS Program are shown in Figure 17A-5. These include the program management practices complemented by those practices for procurement and manufacturing.

The major elements of the BOP Supply Program are shown in Figure 17A-6. These include the program management practices complemented by those for procurement and manufacturing.

The major elements of the NSSS Supplier/RM programs are shown in Figure 17A-7. These include the program management practices complemented by those for design, procurement and manufacturing.

The major elements of the AE program are shown in Figure 17A-8. These include the program management practices complemented by those for design, procurement and manufacturing.

The major elements of the Constructor program are shown in Figure 17A-9. These include the program management practices complemented by those for procurement, manufacturing, construction and installation.

The major elements of the Operator program are shown in Figure 17A-10. These include the program management practices complemented by those for operation, maintenance and modification.

The major elements of the EM program are shown in Figure 17A-11. These include the program management practices complemented by those selected practices associated with design, procurement and manufacturing.

2.3.4 Transfer of Requirements to Others

The delegation of execution responsibility for program elements is accomplished through contracts. These contracts will specify applicable requirements of 10 CFR 50, Appendix B, for contractor quality assurance programs.

The Owner's contracting practice is executed under controlled procedures through which only prospective contractors who are judged to be capable of performing to requisite quality standards are chosen to participate. During the course of their evaluation and selection, the quality achievement and assurance capabilities of prospective contractors are determined and verified through review and inspection and only those with demonstrated capability are awarded contracts. Existing contracts that have been placed will be changed only under controlled procedures.

2.4 SURVEILLANCE

Monitoring of the Project work by Project participants and the quality assurance activities on that work will be performed to verify quality achievement in their work performance. Included in this surveillance function by the Owner is a regular management review practice of the quality assurance program to assess the adequacy of its scope, implementation and effectiveness. Monitoring of the execution of the program will also be performed to verify overall program adequacy.

To accomplish the surveillance function, individuals of the Owner's quality assurance organization are assigned to have cognizance over one or more of the major participant quality assurance programs. Each cognizant individual serves as a focal point to monitor program development and execution, and to coordinate the participant's program with the Owner's program on a day-to-day basis.

The Owner surveillance function is one in which the Project performance and the quality assurance program activities of each contractor are monitored to verify that quality assurance activities are, in fact, meeting their objectives. To implement this function, the Owner has established and implemented procedures:

For monitoring the project work plan as to what activities are being planned, what activities are being achieved, and what activities have been completed and their condition of acceptability.

For monitoring the planning and implementation of first line quality assurance activity which verifies that plans are made and activities are conducted in a timely manner, are meaningful, and which provides verification that the work is actually achieving the quality requirement.

To assure control of approved changes in contracts and other procurement documents.

For assuring that activities related to quality such as inspection and test, are accomplished under suitable controlled conditions including the use of appropriate equipment, a suitable environment such as adequate cleanliness and compliance with necessary prerequisites for the given activity.

For review of contractor or supplier-generated documents for completeness, acceptability and conformance to contract requirements before accepting completed items.

For determining when independent inspections or reviews should be made and conducting those inspections or reviews. To verify that items, documents, services, etc. conform to applicable codes and standards, specifications and other contractual requirements.

For determining the need for an audit of quality assurance activities and scheduling the needed audits.

For maintaining cognizance over the work status and the status of corrective action activities, for determining the effectiveness of the quality assurance activities being performed; and to control and determine disposition of contractor furnished items that do not conform to contract requirements.

These basic elements of surveillance are performed through such activities as the monitoring of work plans, progress and status reports, correspondence, nonconformance reports, inspection reports, audit reports, open item reports, deviation requests and critical item reports, direct surveillance or inspection of work or direct audit of activities, methods and products.

Personnel of the Owner's organization who are assigned responsibility for verifying that contractor performance is in accordance with requirements are selected and assigned to their area of responsibility based upon experience, education, and management's assessment of their performance capabilities. They are observed for performance evaluation on a continuing basis by appropriate management. On-going training and indoctrination programs are conducted to familiarize personnel with technical objectives of the activity being monitored, the requirements that it must meet, the practices and procedures to be executed in verifying conformance to requirements, and the documentation of results.

2.5 INTERFACE COORDINATION

Controls are established to cover interfaces between the physical items of the plant and to cover interfaces between individuals or groups performing other activities as well as quality assurance activities. In general, interface control consists of three elements. These are:

The control of documents through the implementation of document control procedures.

The control of communications using communication control procedures.

The performance of, and participation in, technical reviews such as design reviews.

A management review practice is established through which programmatic interface coordination is accomplished. These reviews are executed as quality assurance program management review meetings which provide a means for assessing Project quality accomplishments, discussing program audits and resolving management problems.

These meetings are held quarterly with a schedule published on a yearly basis. The Owner or his representative chairs each meeting which is attended by management representatives of each major quality assurance program participant. Any program participant may request a program review meeting for any special purpose or problem.

2.6 PROGRAM MANAGEMENT

Quality Assurance Program Management practices, which include those major practices listed below are performed as described in other sections of this program description.

- o Organization
- o Program
- o Planning
- o Document Control
- o Audits
- o Reporting
- o Nonconformance Control
- o Corrective Action
- o Records

The Owner has established a management procedures system in which significant plans and actions are documented including those affecting quality. This system provides a mechanism wherein the policies and objectives of the Project and the Project Quality Assurance Program are defined, documented and promulgated throughout the Owner organization. Within this system detailed procedures for the mandatory actions have been prepared, approved, and issued for use as described in Section 5.1 of this Appendix. These procedures define both the action to be performed and the responsible person or group for performing the action. The Management Procedures Procedure directs implementation of all CRBRP Project Office procedures when approved by the Project Director. To provide positive identification and control of required procedures for quality assurance activities, manuals containing these procedures have been assembled and issued and are closely controlled by the Quality Assurance Division and the Administrative Services Division. These manuals each contain copies of the quality assurance program implementing documents listed in Figures 17A-12 and 17A-13, including a program description from which the program description in this Appendix to Chapter 17 of the PSAR was derived. A brief synopsis of those procedures is contained in Attachment I.

The Quality Assurance Manual is controlled using a document control log which shows the distribution of each copy by copy number including the distribution of revisions. The Quality Engineering Branch of the Quality Assurance Division is responsible for this activity as well as the revision and incorporation of changes to the manual defined and approved by the Chief, Quality Assurance. The contents of the CRBRP Quality Assurance Manual are reviewed and concurrence documented annually as a minimum and the manual is updated as required to maintain it current.

In the execution of the program, should a disagreement arise from a difference of opinion between quality assurance personnel and other Project Office personnel (engineering, procurement, construction, etc.), the principals themselves try to work it out. Should they fail to resolve the differences, the heads of the respective divisions are briefed on the problem by the principals and they attempt to resolve the differences on their level. Should they fail also, the problem is presented to the Project Director by the heads of the Divisions involved, and he arbitrates the matter and renders a decision.

A summary description is provided in Chapter 14 of the PSAR of advanced planning for the control of management and technical interfaces between the Constructor, A/E, NSSS Vendor, and Owner during the phaseout of design and construction and during preoperational testing and plant turnover.

3.0 DESIGN CONTROL

3.1 OWNER IMPLEMENTATION

The Owner performs no design in the Project but has assigned the design responsibility to other major participants in the Project. The Owner has established and specified the design guidelines for the CRBRP. In addition, the Owner has established and specified the essential major plant parameters to be incorporated into the design by the cognizant design participants. The Owner monitors the development of the plant design through participation in design planning, review, and development meetings with the appropriate design organizations.

The Owner has established a Project-wide design control system that is based primarily on a family of System Design Descriptions (SDD) as the major vehicle for design documentation, review and approval. Through the SDD, the cognizant design contractors are responsible for the development and definitizing of the Plant systems and components design for which they are assigned responsibility. All SDDs are approved by the Owner. The Owner, therefore, maintains the technical supervision and administration of overall aspects of the CRBRP design. As such, the Owner has the responsibility, authority, and accountability for all aspects of the CRBRP design and design control within the specified design, cost and schedule constraints for the Project.

To implement the design control function, the Owner has established design review and approval requirements based upon a four-level classification system as listed below, and has provided for both external and internal design interface controls. Externally, the Owner has defined interfaces and provides direction to the responsible design organizations as follows:

Type 1 Data - Requires Owner Approval
Type 2 Data - Requires NSSS Supplier and/or AE Approval
Type 3 Data - Requires RM or AE Approval
Type 4 Data - May be Supplier Approved

- NSSS Supplier - The Owner provides direction directly to the NSSS Supplier. For systems and equipment for which the NSSS Supplier has design responsiblity, he recommends approval of Type 1 Data directly to the Owner and the Owner takes approval action. The NSSS Supplier provides Type 2, 3, and 4 data to the Owner for information and the Owner reviews and takes action only as appropriate.

- Reactor Manufacturer - The Owner provides direction to the Reactor Manufacturers (RMs), who are major subcontractors to the NSSS Supplier, through the NSSS Supplier for Type 1 systems and equipment data. The RMs submit these data to the NSSS Supplier and the Owner in parallel, and request approval from the NSSS Supplier. The NSSS Supplier in turn reviews and comments on these data, and submits a recommendation to the Owner for approval. The Owner approves or disapproves after receipt of the recommendation. The RM initiates implementation of the activity upon receipt of the Owner approval. In parallel, the NSSS Supplier issues a confirmatory authorization for the RM to proceed on the basis of the Owner Approval Action. The RM provides Type 2, 3, and 4 Data to the Owner for information and the Owner reviews and takes action only as appropriate. (Normally, no Owner action is required).

- Architect-Engineer - The Owner provides direction directly to the Architect-Engineer (AE). For systems and equipment for which the AE has design responsibility, the AE recommends approval of Type 1 data directly to the Owner and the Owner takes approval action. The AE provides Type 2, 3, and 4 data to the Owner for information and the Owner reviews and takes action only as appropriate.

The Owner has established an overall Design Interface Control System for the Project. The system provides for control of system and equipment functional, parametric, and physical interface requirements for all portions of the CRBRP. Each design contractor is required to assure the accuracy and completeness of interface data pertaining to systems and equipment under his cognizance.

Functional and parametric interfaces are controlled through the SDDs. Physical interfaces are controlled by Interface Control Drawings (ICD) or appropriate vendor documentation. The ICDs identify physical interface characteristics that are necessary to ensure compatability geometrically, functionally, and with processes and environment between the overall equipment and its surrounding structure/facility (including mechanical, electrical and fluid requirements).

In the Project Design Interface Control System the NSSS Supplier is responsible for assuring that system-to-system interfaces are established and maintained current and for approving ICDs. The NSSS Supplier is responsible for assuring that the required System-to-Structure/Facility interface data are provided to the AE and for approval of its use.

The AE is responsible to identify and schedule needed equipment to structure/facility interface data for design of interfacing systems, equipment and structures under its cognizance. The AE has the same responsibility for BOP system-to-system interfaces as described above for the NSSS Supplier.

Each design contractor is responsible for managing, scheduling, preparing, maintaining, and updating interface data and applicable ICDs for mating equipment, systems and structures under his cognizance.

The responsibility for performing interface coordination is assigned to the cognizant engineers of the responsible design organizations. The cognizant engineer assures that:

a) Adequate interface control is exercised for his area of cognizance.

b) ICDs for his system interfaces are identified, entered into the Project documentation status reporting system and produced.

c) That ICDs reflect up-to-date information, and that system/component designs within his scope of responsibility remains within constraints of approved SDDs and ICDs.

d) Interface requirements between equipment and other equipment/facilities and/or structures, if appropriate, is included in the equipment specifications.

e) That all "HOLDS" that impose a constraint upon an activity are properly identified on the ICD and that these "HOLDS" are prepared for input into the Project HOLD status system.

f) That all ICDs are approved in accordance with the approval requirement for Principal Design Data.

g) That changes required to ICDs are processed in accordance with Change Control requirements.

The responsible cognizant engineer provides an up-to-date list of interface requirements to other affected cognizant engineers and coordinates proposed changes with them prior to implementation.

A supplier who is responsible for component design will be required to develop interface requirements that his component imposes upon any mating equipment. The supplier is or will be required to provide this data to the responsible design contractor for use in developing the required interface.

Internally, the Owner exercises control over the CRBRP design by three basic processes.

a. Review and approval of Principal Design Data and significant changes of these data including the verification of design documents such as SDDs, design input and criteria design drawings, design analyses, computer programs, specifications and procedures.

b. Performance of Design reviews and overview of contractor Design Review.

c. Technical Direction - In addition to scheduled reviews and approvals (items a and b above) the Owner exercises design control by continual evaluation of design contractors' technical accomplishments on a day-to-day working level basis and by providing technical guidance as necessary to meet Project technical, schedular and cost objectives. This guidance may be provided by memorandum from the Owner, by meeting agreements, which are confirmed for implementation in writing by the Owner, or by telephone conversations that are subsequently confirmed in writing. Cognizant engineers maintain continuous contact with cognizant contractor and field office counterparts to keep informed on status of work, actions being taken, and other significant activities.

The review and approval of principal design data and the performance of design reviews include cognizant personnel of the Engineering, Quality Assurance, and Public Safety Divisions of the Project Office and the use and approval of appropriate design review checklists to verify that:

1. Suitable design controls are applied to such disciplines as reactor physics; seismic, stress, thermal, hydraulic, radiation, and the SAR accident analyses associated computer programs, compatibility of materials; and accessibility for inservice inspection, maintenance and repair.

2. Proper selection and accomplishment of design verification or checking methods such as design reviews, alternate calculations to include the development, control and use of computer code programs, or qualification testing have been performed by the cognizant design contractors.

3. When a test program is used to verify the adequacy of a design, a qualification test of a prototype unit under adverse design conditions is used.

4. Individuals or groups that were responsible for design verification were other than the original designer and the designer's immediate supervisor.

5. The applicable regulatory requirements and design bases are correctly translated into the specifications and drawings and that deviations from specified quality standards are identified and procedures are established to ensure their control.

6. Appropriate and required quality standards are specified in the design documents.

7. Design and specification changes, including field changes were subjected to the same design controls and approvals that were applicable to the original design.

8. Standard "off the shelf" commercial or previously approved materials, parts, and equipment that are essential to the safety related functions of structures, systems, and components are reviewed for suitability of application prior to selection.

9. Designs are reviewed to assure that design characteristics can be controlled, inspected and tested, and inspection and test data are identified.

10. The selection of suitable computer programs, materials, parts, equipment, and processes for safety-related structures, systems, and components has included the use of authorized computer programs, valid industry standards and specifications, material and prototype hardware testing programs, and design reviews.

The Owner receives these documents into an engineering document control system which insures accountability throughout the review process. Each design document is assigned to a cognizant engineer of the Owner engineering organization who reviews the design for adequacy and coordinates that document through the review process. He selects the additional reviewers necessary and routes these documents through quality assurance for further review. These design documents and their revisions are distributed in a timely manner and the control system is so designed to prevent inadvertent use of superseded material. The cognizant engineer serves as a monitor for this process and also assures that errors and deficiencies that adversely affect safety-related structures, systems and components in the design process, including design methods (such as computer codes) are documented, reported according to code, standard or regulation and followed up to see that appropriate corrective action is taken. The cognizant engineer coordinates the resulting comments and forwards them through top management to the originator.

Changes in design documentation, including field changes, are processed with the same requirements as the original design. The Owner monitors the design control measures exercised through surveillance and audit as part of the quality assurance program. Design documents, review records, and changes thereto are collected, stored and maintained in a systematic and controlled manner through the Owner design document control and quality records activities. These activities as well as the entire CRBRP design control process are covered by written procedures and are executed to carry out design activities in a planned, controlled and orderly manner.

The Owner has established a Project-wide Design Change Control practice that requires that changes to approved design data are processed as an Engineering Change Proposal (ECP) receiving the required approvals based upon the Data Type involved. Once the change has been approved, documentation is issued that defines and authorizes the change and directs implementation of the change.

In addition to the activities already outlined for technical direction, the Owner reviews and accepts for Project use the programmatic documentation which affects the design control activities. Owner procedures have been implemented which assure that the applicable regulatory requirements and design bases are correctly translated in the written procedures and instructions of contractor programs as well as within the Owner's internal program. These procedures also assure that changes to requirements and quality standards are controlled and are properly implemented.

The Project design guidelines were prepared, documented and issued under a closely controlled process. These guidelines are maintained by that same controlled process in accordance with established procedures. The Owner quality assurance organization participates in and monitors the design control practice and periodically audits this practice to assure implementation and adequacy.

3.2 REQUIREMENTS OF OTHER PARTICIPANTS

CRBRP Project participants who are assigned design responsibilities are required by contract to exercise design control practices in accordance with specified requirements. These practices include the following:

- o Design Planning

- o Design Definition and Control

  1. Design Criteria
  2. Codes, Standards and Practices
  3. Engineering Studies and Analyses
  4. Parts, Materials and Processes
  5. Design Descriptions
  6. Specifications, Drawings and Instructions
  7. Identification
  8. Acceptance Criteria
  9. Interface Control
  10. Engineering Holds
  11. Calculations
  12. Computer Codes

- o Document Review and Control

  1. Document Reviews
  2. Document Control
  3. Engineering Drawing Lists
  4. Drawing Checks

o Design Reviews

Required design verification for the level of design activity accomplished is to be performed prior to release for procurement, manufacture, construction, or release to another organization for use in other design activities. In all cases, the design verification is to be completed prior to relying upon the component, system, or structure to perform its function.

o Development

(Including provisions that prototype component or feature testing is performed as early as possible prior to installation of plant equipment or prior to the point when the installation would become irreversible.)

o Failure Reporting and Corrective Action

The required practices are to include the review of design drawings and specifications by the QA organization to assure that the documents are prepared, reviewed, and approved in accordance with internal procedures and that the documents contain the necessary quality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

The Owner monitors major participant design control practices and periodically audits the participants practices to specified requirements to assure proper implementation and adequacy of the practice.

4.0 PROCUREMENT DOCUMENT CONTROL

4.1 OWNER IMPLEMENTATION

The Owner does not normally procure plant components directly, but does so through other Project participants. However, when the Owner does choose to procure plant components directly, it accomplishes this action in accordance with the same requirements as imposed on other Project participants. The Owner has established and implemented a practice for control of procurement documents to assure that procurement functions are accomplished in accordance with the applicable contracts, codes, standards, drawings, and specifications. This practice is carried out under written procedures which provide for coordination and implementation of procurement planning, procurement activities among Project participants and review of procurement documents such as preprocurement plans and purchase orders, and changes and/or modifications thereto by designated personnel to assure that these documents are complete and correct.

Technical portions of procurement documents are prepared by the AE, RM's and the Constructor, approved by the NSSS Supplier for RM items and submitted to the Owner for review and approval. When a plant procurement document is received by the Owner, it is routed to the Assistant Director for Procurement (AD/PR). The AD/PR coordinates the Owner review in conjunction with the cognizant engineer. The review is conducted thoroughly but as promptly as possible. To conduct reviews and expedite approval of procurement documents, a checklist is used by procurement document reviewers. These checklists are rather extensive and include such check points as whether or not the procurement documents:

1. Identify the documentation (e.g., drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and material chemical and physical test results) to be prepared, maintained, and submitted as applicable to the purchaser for review and approval.

2. Contain or reference the design basis technical requirements including the applicable regulatory requirements, components, and material identification requirements, drawings, specification, codes and industrial standards, test and inspection requirements, and special process instructions for such activities as welding, heat treating, nondestructive testing, and cleaning.

3. Identify the applicable 10 CFR 50, Appendix B, requirements which must be complied with and described in the supplier's QA program.

4. Identify those records which shall be retained, controlled, maintained, or delivered to the purchaser prior to use or installation of the hardware.

5. Contain the procuring agency's right of access to supplier's facilities and records for source inspection and audit.

6. Provide for spare or replacement parts of safety-related structures, systems, and components being subject to controls at least equivalent to those used for the original equipment.

All changes and revisions to procurement documents are subject to the same review and approval requirements as the original documents.

Procurement document reviewers are specified by the AD/PR and the cognizant engineer. Reviewers are selected on the basis of their qualifications and their ability to provide a meaningful input to a particular document. The selected Owner reviewers include as a minimum, the AD/PR or his designated representative, the head of the Owner quality assurance organization or his designated representative and the Owner cognizant engineer. The selected reviewers also include contract and procurement staff and, as appropriate counsel, patent and finance representatives. The respective cognizant engineer has the principal responsibility for determining the suitability and adequacy of technical specifications included in procurements documents.

The quality assurance reviewer has the principal responsibility for determining the adequacy of the procurement documents regarding quality assurance requirements. He is trained and qualified in quality assurance practices and concepts to make this determination. This review is to determine that quality and quality assurance requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements.

The AD/PR has the principal responsibility for determining the overall adequacy of administrative, financial, and contractual aspects of procurement documents.

Comments from reviewers of procurement documents are made in writing. After receiving and resolving all comments, the AD/PR prepares formal correspondence to the appropriate participant and reflects comments, approval, or notifies the participant of the reasons for disapproval.

The Owner quality assurance organization both participates in and monitors the execution of this practice. Periodically the Owner quality assurance organization audits or arranges for independent audit of this practice to assure implementation and adequacy.

4.2 REQUIREMENTS OF PARTICIPANTS

Each Project participant, who is assigned procurement responsibility is required by contract to implement and maintain a procurement document control practice that fulfills the assigned quality assurance requirements. This practice will include the preparation of procurement documents to contain the following:

- Scope of Work
- Technical Requirements
- Quality Assurance Program Requirements
- Right of Access
- Special Quality Assurance Requirements
- Documentation Requirements
- Nonconformance Control
- Transfer of Requirements to Lower Tier Participants

This practice will also include:

- Procurement Document Review and Approval
- Document Control (Release, Distribution and Change)

The Owner monitors major participant procurement document control practices and periodically audits the participants practice to assure proper implementation and adequacy.

5.0 INSTRUCTIONS, PROCEDURES AND DRAWINGS

5.1 OWNER IMPLEMENTATION

The Owner has established and implemented a practice of prescribing in documentary form the required quality of plant structures, systems, and components and necessary activities to assure attainment of requisite quality through work activities. This practice includes specifying division of work responsibilities and the Project-wide practices to be implemented in execution of those responsibilities. Through this practice, the following documents have been prepared:

- o Plant Design Guidelines
- o Management Policies and Requirements
- o Contract Statements of Work
- o Environmental and Safety Analysis Reports
- o Policies, Procedures and Instructions
- o Reports
- o Records

The Owner has prepared his procedures and instructions in accordance with procedures that prescribe the format to be followed and the identification system to be used. These procedures cover all activities of management, engineering and design control, document review and control, procurement, surveillance activities, audits, and records management. These procedures prescribe methods for performing quality-related activities in conformance with the requirements of 10 CFR 50, Appendix B.

The Owner procedures are organized under a Management Procedures System which is administered by a procedures coordinator from within the Owner organization. The procedures coordinator is assigned the function of controlling the issuance of procedures to assure coordination and consistency in format, content, etc. The procedures system itself is organized along divisional lines (Engineering, Procurement, Construction, Quality Assurance, Public Safety, Operations, Project Control, Administrative Services, and others) which give the responsible managers the responsibilities for:

- o Assuring that policies of a continuing nature are incorporated in the Management Procedures System (MPS).
- o Incorporating applicable laws, standards such as 10 CFR 50, Appendix B, Executive Orders, decisions and directives of the Project Steering Committee (PSC) into the procedures to the extent necessary to show the requirements placed upon the Owner.
- o Determining the coverage and content of management directives necessary to carry out their assigned functions, assuring the accuracy and currency of the procedures, and arranging for the cancellation of those that become obsolete.
- o Approving procedures for which they are responsible. Obtaining review, comment, and document concurrences by other organizational units when appropriate.
- o Submitting to the Procedures Coordinator:

  a. Draft Procedures for Review of Format
  b. Final Procedures for Director Approval

- o Determining, with concurrence of General Counsel, what portions of procedures, if any, shall be communicated to the contractors. Furnishing to the Procedures Coordinator the names of contractor personnel to whom such material together with any appropriate supplementary explanation or instructions should be distributed.

The Procedures Coordinator: assures that style, format, content, terms, titles and numbering sequence of all procedures conform to the requirements of the Management Procedures System.

The Chief, Administrative Services, is the prime control officer for procedures and as such:

- o Effects the printing and distribution of the final approved procedures and subsequent revisions.

- o Maintains a master file of all current approved CRBRP Project Office procedures and a reference file of previously issued procedures and their revision.

- o Prepares and maintains an index of procedures.

Organization Unit Managers are responsible for writing and implementing the procedures necessary for their division. General Administration procedures cover policies and procedures which apply to all employees. The Project Director approves for issuance all CRBRP Project Office procedures. The individual division procedures are approved by the responsible Division Manager and recommended to the CRBRP Project Director for final approval.

Each new procedure or revision of existing procedure is prepared using the Management Procedures System numbering code and format.

Each division established steps for the review of draft procedures within the division. If a procedure applies to more than one division, the other divisions affected receive the draft procedure for review. A draft is sent to the Procedure Coordinator who reviews it for format, style, and numbering sequence.

| The final procedure or revision of an existing procedure is approved by the appropriate division manager responsible for that particular subdivision of
| procedures, is concurred in by the Chief, Quality Assurance, and is approved by the CRBRP Project Director, and released for implementation.

Distribution of each procedure or revision of existing procedure is listed and filed with the procedure in the procedure master file. The register shows which revision is current.

The Owner's practice for documenting, in written form, the requirements for and results of activities affecting quality is, itself, executed in accordance with document control procedures identified under Section 6.0, Document Control.

The Owner's Quality Assurance Organization both participantes in and monitors the execution of this practice. Periodically the Quality Assurance organization audits or arranges for independent audit of this practice to assure implementation and adequacy.

5.2 REQUIREMENTS OF OTHER PARTICIPANTS

CRBRP Project participants, who are assigned responsibility for performing work activities affecting quality, are required by contract to establish and implement a practice of prescribing those activities in documented form that fulfills the quality assurance requirements. This practice will include the preparation of the following types of documents:

- o Policies, Procedures and Instructions
- o Quality Records
- o Quality Status Reports
- o Design Specifications to include Quantitative Acceptance Criteria such as Dimensions, Tolerances, and Operating Limits
- o Design, Manufacturing, Construction and Installation Drawings to include as-built drawings that accurately reflect the actual plant configuration
- o System Design Descriptions
- o Manufacturing, Construction, Installation, Inspection and Testing Specifications, Procedures, and Instructions to include Qualitative Acceptance Criteria such as workmanship samples
- o Test Procedures
- o Topical Reports and input to SARs

The Owner monitors major participant documentation practices and periodically audits the participants practice to assure proper implementation and adequacy and to verify that important activities have been satisfactorily accomplished.

6.0 DOCUMENT CONTROL

6.1 OWNER IMPLEMENTATION

The Owner has established and implemented a document control system that fulfills the quality assurance program requirements and applies to those types of documents prepared by the Owner and identified in Sections 3, 4 and 5 of this description.

The controlled documents originated by the Owner are processed in a controlled manner to assure the following:

- o Uniformity of format of initial and subsequent issuances.

- o Proper identification as to the originator and date of origin of a document, and a mechanism for verification of the authenticity of information.

- o Positive review and approval by persons qualified to determine the correctness of the information presented and to judge its ultimate usefulness.

- o Prompt and accurate issuance and distribution, including a mechanism for receipt control, of both the original document and subsequent revisions to prevent inadvertent use of superseded material and to place documents in work areas in a timely manner.

- o Efficient revision of documents when necessary to clarify, correct, augment or up-date the content of a document, while preserving the integrity of originally approved and released information.

- o Quality Assurance requirements are properly stated, are adequate and are included prior to implementation.

- o Documents are available at the location where the activity will be performed prior to commencing the work.

Controlled documents are standardized by procedure as to identification, format, and numbering. These documents are reviewed for adequacy by Division Chiefs, and/or the CRBRP Project Director, as appropriate. The Chief of the Division originating the controlled document determines the extent of necessary reviews. The draft controlled document is routed to the appropriate reviewing personnel/organizations. Comments of reviewing personnel are resolved prior to final approval of the document. A record of the review sequence which has been accomplished is documented and retained. Changes or revisions are reviewed and approved by the same Divisions that performed the original review and approval. If the controlled document will be issued only to personnel of the originating Division, the respective Division Chief may approve the document for issue upon completion of necessary reviews. If the controlled document is to be issued to personnel outside of the originating Division, the respective Division Chief secures any necessary higher level approvals. The Chief of the Division originating a controlled document established an appropriate periodic review schedule for the approved document. The primary purpose of these reviews is to determine if changes in Project status have resulted in the need for revisions to the controlled documents.

The originating Division establishes and maintains an appropriate listing of the distribution of the document upon issue. A receipt page is attached to the transmitted controlled document which requests the person receiving the document to sign and date the page and return it to the originating Division. A designated person initials the respective distribution listing upon receipt of the signed page to reflect accomplishment of transmittal and receipt. He also reviews the Division's controlled document distribution listing at least bi-monthly to follow-up on any delinquent receipt pages. This distribution listing is a master list which is updated periodically to show current revision, number distributed, location, etc. Revisions to controlled documents are systematically processed with the same procedure as the original. Changes are also reviewed and approved by the same Divisions that performed the original review and approvals.

The Owner Quality Assurance organization both participates in and monitors the execution of the Document Control System. Periodically the Quality Assurance Organization audits or arranges for independent audit of the Document control System to assure implementation and adequacy.

6.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each Project participant is required to implement and maintain a Document Control System that fulfills the assigned requirements imposed by contract and to control these documents prepared as described in Section 5.2 of this description. The Document Control System will include elements of:

- o Document Preparation
- o Document Identification
- o Document Review and Approval
- o Document Issue, Distribution and Storage
- o Document Revision

The Owner monitors major participant Document Control Systems and periodically audits the participants system to assure proper implementation and adequacy.

7.0 CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICES

7.1 OWNER IMPLEMENTATION

The Owner has established a practice for control of contracting activities associated with the major contractors participating in the CRBRP Project, including the direct procurement of components when the Owner chooses to be the purchaser. The contracting practice includes the following:

- o Contract Document Preparation, Review and Change Control
- o Selection of Contractors

  1. Qualified personnel evaluate the contractors capability to provide services and products of acceptable quality.
  2. The Quality Assurance, Engineering and Procurement personnel participate in the evaluation of contractors.
  3. The evaluation of contractors include one or more of the following:

     a. The contractor's capability to comply with the elements of 10 CFR 50, Appendix B which are applicable to the defined scope of work.

b. A review of previous records and performance of contractors who have provided a similar scope of work to others.

c. A survey of the contractor's facilities and quality assurance program practices to determine his capability to perform the required scope of work in accordance with requirements. Results of these surveys are documented and filed in the Owner's office.

o Proposal Evaluation and Award

o Customer's control of contractor's performance.

1. Surveillance of contractors during design, procurement, maintenance, installation and operation of materials, equipment, components and facilities is planned and performed with quality assurance organization participation in accordance with written procedures to assure conformance to specified requirements. These procedures provide for:

   a. Specification of the activities to be monitored or otherwise verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

   b. Audits and surveillance that verifies contractor compliance with quality and quality assurance requirement. Surveillance is performed on an on-going basis to verify acceptable performance.

o Acceptance of item or service by or for the customer

1. Activities at point of receipt of supplier-furnished material, equipment and services is performed to assure that:

   a. The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and with the receiving documentation.

   b. Inspection of the material, component or equipment, and acceptance records is performed and judged acceptable in accordance with predetermined inspection instructions, prior to installation of use.

   c. Inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment are available at plant prior to installation or use.

   d. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

e. Nonconforming items are segregated when practicable, controlled, and clearly identified until proper disposition is made.

o Corrective Action with regard to the contracting process

o Quality Assurance Records

1. The contractor furnishes the following records, as a minimum, to the customer.

   a. Certifications that specifically identify (e.g, by the purchase order number) the purchased material or equipment and the specific procurement requirements (codes, standards, specifications, etc.) met by the items.

   b. Certifications that identify any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair."

Review and approval of data by qualified personnel is performed in accordance with established procedures as described in Section 3.0 and 6.0 of this Appendix.

Procurement of spare or replacement parts for structures, systems, and components important to safety is subject to present quality assurance program controls, to codes and standards, and to technical requirements equal to or better than the original technical requirements, or as required to preclude repetition of defects.

7.2 REQUIREMENTS OF OTHER PARTICIPANTS

Project participants, who are assigned responsibility for procurement of material, equipment and services, are required, by contract, to establish and implement a system for control of those procurements including interfaces between design, procurement, and quality assurance organizations. Each system will include the following elements:

o Procurement Document Preparation, Review and Change Control

o Selection of Procurement Sources

1. Qualified personnel evaluate the supplier's capability to provide services and products of acceptable quality.

2. The Quality Assurance and Engineering personnel participate in the evaluation of those suppliers providing critical items.

3. The evaluation of suppliers is based on one or more of the following:

   a. The supplier's capability to comply with the elements of 10 CFR 50, Appendix B, which are applicable to the type of material, equipment, or service being procured.

b. A review of previous records and performance of suppliers who have provided similar articles of the type being procured.

c. A survey of the supplier's facilities and quality assurance program practices to determine his capability to supply a product which meets the design, manufacturing, and quality assurance requirements. Results of these surveys are to be documented and filed at the buyer's facility. If an LCVIP letter of confirmation or the "CASE" Register is used to establish the qualifications of the supplier, the documentation on file is to identify the letter or audit used.

o Bid-Evaluation and Award

o Purchaser's control of supplier's performance.

1. Surveillance of suppliers during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed with quality assurance organization participation in accordance with written procedures to assure conformance to the purchase order requirements. These procedures provide for:

a. Specification of the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits and surveillance which assure that the supplier complies with all quality and quality assurance requirements. Surveillance is performed on those items where verification of conformance to procurement requirements cannot be determined upon receipt.

c. Special quality verification activities to provide the necessary assurance of an acceptable item for commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed by the purchaser in a practicable manner.

o Acceptance of item or service

1. Receiving activities associated with supplier-furnished material, equipment and services is performed to assure that:

a. The material, component, or equipment, is properly identified and corresponds (to the identification on the purchase document and) with the receiving documentation.

b. Inspection of the material, component or equipment, and acceptance records is performed and judged acceptable in accordance with predetermined inspection instructions, prior to installation or use.

c. Inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment are available at plant site prior to installation or use.

d. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

e. Nonconforming items are segregated when practicable, controlled, and clearly identified until proper disposition is made.

o Corrective action with regard to the procurement process

o Quality assurance records

1. The supplier furnishes the following records, as a minimum, to the purchaser.

   a. Certifications that specifically identify (e.g., by the purchase order number) the purchased material or equipment and the specific procurement requirements (code, standards, specifications, etc.) met by the items.

   b. Certifications that identify any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair."

The review and approval of supplier-furnished data by qualified personnel is performed in accordance with established procedures.

Each major participant reviews and provides appropriate approval of supplier generated documentation, such as certifications, for completeness, acceptability, and conformance to contract requirements before accepting completed items. At receipt inspection, each contractor routinely or periodically validates supplier-furnished material certifications by means of independent analysis or overchecks. The participants receipt inspection planning defines the necessary inspections and tests and provides for inspection density adjustment depending upon source, quality performance history, lot size and other factors.

| The Owner monitors major participant procurement practices and periodically audits the participant's practices to assure proper implementation and adequacy.

8.0 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

8.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for identification and control of materials, parts and components to the other major project participants through contracts.

8.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each project participant, who has an assigned responsibility for materials (including consumables), parts and components including partially fabricated subassemblies is required, by contract, to establish and implement identification and control practices. The description of the practices should include organizational responsibilities. Each participant's identification requirements are to be determined during the initial planning stages and their practice will assure: that identification of the item is maintained, both on or attached to the item and on records traceable to the item as required throughout fabrication, erection, installation, and use of the item; the item(s) can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical mill test reports; that the method and location of identification does not affect the function or quality of the item being identified; and that the correct identification of items is accomplished and verified prior to the release for fabrication, assembly, shipping and installation. These practices will be designed to preclude the use of incorrect or defective materials, and parts important to the function of safety-related structures, systems, and components.

The Owner monitors major participant identification and control of materials, parts and components practices and periodically audits the participants' practices to assure proper implementation and adequacy.

9.0 CONTROL OF SPECIAL PROCESSES

9.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for control of special processes to the other major Project participants through contracts.

9.2 REQUIREMENTS OF OTHER PARTICIPANTS

Project participants who are assigned responsibility for activities where special processes are involved, are required, by contract, to establish and implement practices to assure adequate performance and control of special processes such as welding, heat treating, nondestructive examination, bending, coating, soldering, plating, hard surfacing and cleaning. These practices will include the following elements:

- o Description of the criteria for determining those processes that are controlled as special processes. A listing of special processes, which are generally those processes where direct inspection is impossible or disadvantageous, should be provided.

- o Description of the organizational responsibilities, including those for the quality assurance organization, for qualification of special processes, equipment, and personnel.

- o Qualification of procedures, equipment, and personnel for performance of special processes in accordance with applicable codes, standards, quality assurance procedures, specifications, or supplementary requirements. The quality assurance organization is involved in the qualification activities to assure that they are satisfactorily performed.

- o Special processes are performed by qualified personnel using qualified procedures and equipment with established procedures for recording evidence of acceptable accomplishment.

- o Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

The Owner monitors major participant special processes control practices and periodically audits the participants' practices to assure proper implementation and adequacy.

10.0 INSPECTION

10.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for direct inspection of items and work practices to each major participant in his assigned scope of work by contract. The Owner or his agent may perform limited inspection and witnesses inspections on a selective basis. When the Owner elects to perform or have his agent perform inspections, the practice followed will conform to the same requirements imposed upon other participants.

10.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each participant who is assigned responsibility for performing procurement, manufacturing, construction, or operation activities that affect quality, is required, by contract, to establish and implement an inspection practice of sufficient scope to be fully effective. The inspection practice will identify and verify conformance of items and services with the documented specifications, instructions, procedures and drawings for accomplishing the required activities. The program will assure that:

- o Organizational responsibilities for inspection are described. Inspection personnel are independent from the individual or group performing the activity being inspected. If the individuals performing inspections are not part of the quality assurance organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure such as cost and schedule should be reviewed and found acceptable by the quality assurance organization prior to initiation of the activity.

- o Program procedures are developed with quality assurance organization participation to provide criteria for determining the accuracy requirements of inspection equipment and the timing of required inspection, or definition of how and when inspections are performed.

- o Inspection procedures, instructions, and checklists contain the following:

a. Identification of characteristics to be inspected.

b. Identification of the individuals or groups responsible for performing the inspection operations.

c. Acceptance and rejection criteria.

d. A description of the method of inspection.

e. Specification of the necessary measuring and test equipment, including accuracy requirements.

f. Identification of required procedures, drawings and specifications, including applicable edition or revision.

- Inspection procedures or instructions are available with necessary drawings and specifications for use prior to performing inspection operations.

- Inspectors (including NDT personnel) are qualified in accordance with appropriate codes, standards, and company training programs, and their qualifications and qualification records (certifications) are kept current.

- Modifications, repairs, and replacements are inspected in accordance with the original design and inspection requirements or acceptable alternatives.

- Provisions are established that identify mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

- The individuals or groups who perform receiving and process verification inspections are identified.

- Provisions are established for indirect control by monitoring processing methods, equipment, and personnel if direct inspection is not possible.

- Inspection results are documented, evaluated and their acceptability determined by a responsible individual or group.

The Owner monitors major participant inspection practices and periodically audits the participants practice to assure proper implementation and adequacy.

11.0 TEST CONTROL

11.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for testing and test control practices to the other major participants by contracts. The Owner will witness tests by the major project participants on a selective basis and he or his agent may, upon limited occasions, conduct independent tests of his own. When the Owner elects to perform or have his agent perform tests, the test control practices will conform to the same requirements imposed upon other participants.

11.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each participant, who is assigned responsibilities for performing development, procurement, manufacturing, construction or operation activities that affect quality, is required by contract to establish required tests, including proof tests prior to installation and preoperational tests and to establish a testing control practice which will include the following elements:

- o Identification of required testing to demonstrate that the item will perform satisfactorily in service and that testing activities are identified, documented, and accomplished in accordance with written controlled procedures.
- o Written test procedures that incorporate or reference the requirements and acceptance limits contained in applicable design and procurement documents.
- o Written test procedures that include:
  - a. Instructions for testing method and test equipment and instrumentation.
  - b. Provisions for the following as appropriate:
    - Calibrated instrumentation
    - Adequate and appropriate equipment
    - Trained, qualified, and licensed or certified personnel
    - Preparation, condition, and completeness of item to be tested
    - Suitable and controlled environmental conditions
    - Mandatory inspection hold points for witness by Owner, contractor, or authorized inspector
    - Provisions for data collection and storage
    - Acceptance and rejection criteria
    - Methods of documenting and recording test data and results
    - Provisions for assuring test prerequisites have been met
- o Test results are documented, evaluated, and acceptance status identified by a qualified, responsible individual or group.
- o Modifications, repairs, and replacements are tested in accordance with the original design and testing requirements or acceptable alternates.

The Owner monitors major participant testing and test control and periodically audits the participants' test control practices to assure implementation and adequacy.

12.0 CONTROL OF MEASURING AND TEST EQUIPMENT

12.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for control of measuring and test equipment to the other major Project participants through contract. In those instances where the Owner elects to perform or have his agent perform inspections, examinations, or tests, the measuring and test equipment will be controlled in accordance with those requirements imposed upon the participants.

12.2 REQUIREMENTS OF OTHER PARTICIPANTS

Project participants, who are assigned responsibility for performing inspections, examinations, or tests, are required by contract to establish and implement a system for calibration and control of measuring and test equipment.

This system will include the following elements:

- o Procedures that describe the calibration technique and frequency, maintenance, and control of all measuring and test instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment which are used in the measurement, inspection, and monitoring of safety-related components, systems, and structures. The review and documented concurrence of these procedures is described and the organizations responsible for these functions are identified.

- o Measuring and test equipment is identified and the calibration test data is identified as to the equipment to which it applies.

- o Measuring and test equipment is labeled or tagged or "otherwise controlled" to indicate the due date of the next calibration. The method of "otherwise controlled" shall be described.

- o Measuring and test instruments are calibrated at specified intervals based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement.

- o An investigation conducted and documented to determine the validity of previous inspections performed when measuring and test equipment is found to be out of calibration.

- o Calibrating standards have an uncertainty (error) requirement of one-fourth to one-tenth of the tolerance of the equipment being calibrated. A greater uncertainty may be acceptable when limited by the "state-of-the-art." Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if it can be shown to be adequate for the requirements and the basis of acceptance is documented and authorized by responsible management. The management authorized to perform this function is identified.

- Measures are taken and documented to determine the validity of previous inspections performed and the acceptability of items inspected or tested since the last calibration when measuring and test equipment is found to be out of calibration. Inspections or tests are repeated on items determined to be suspect.

- Reference and transfer standards are traceable to nationally recognized standards; or, where national standards do not exist, provisions are established to document the basis for calibration.

The Owner monitors major participants calibration and control of measuring and test equipment and periodically audits the participants' practices to assure implementation and adequacy.

13.0 HANDLING, STORAGE AND SHIPPING

13.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for handling, storage and shipping practices to the other major participants through contracts.

13.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each participant, who is assigned responsibility for manufacturing, fabrication or assembly, is required by contract to establish and implement practices for handling, storage and shipping of items. These practices will include the following:

- Special handling, preservation, storage, cleaning, packaging, and shipping requirements are specified and accomplished by suitably qualified individuals in accordance with predetermined work and inspection instructions.

- Procedures are prepared in accordance with design and procurement specification requirements to establish and describe controls for the cleaning, handling, storage, packaging, shipping, and preservation of materials, components and systems to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity.

The Owner monitors major participant handling, storage and shipping practices and periodically audits participant practices to assure implementation and adequacy.

14.0 INSPECTION, TEST AND OPERATING STATUS

14.1 OWNER IMPLEMENTATION

The Owner delegates execution responsibility for inspection, test and operating status measures to the other major participants through contracts.

14.2 REQUIREMENTS OF OTHER PARTICIPANTS

Participants, who are assigned responsibility for manufacturing, construction, or operation, are required to establish and implement practices to indicate the status of inspections and tests performed upon individual items throughout fabrication, installation and test by using such markings as stamps, tags, labels, routing cards or other suitable means. These practices will include provisions for:

- o Identification of the inspection, test, and operating status of structures, systems, and components being known throughout manufacturing and installation.
- o The application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps being controlled.
- o Altering the sequence of or bypassing the required inspections, tests, and other critical or safety-related operations being controlled through documented measures under the cognizance of the QA organization. Such actions are to be subject to the same controls as the original review and approval.
- o The status of nonconforming, inoperative, or malfunctioning structures, systems, or components being documented and identified to prevent inadvertent use. The organization responsible for this function is identified.

The Owner monitors major participant practices for indicating inspection, test and operating status and periodically audits participant practices to assure implementation and adequacy.

15.0 NONCONFORMING MATERIALS, PARTS OR COMPONENTS

15.1 OWNER IMPLEMENTATION

The Owner has established and implemented practices for control, review and disposition of nonconforming materials, parts or components. These practices are designed to assure that measures are established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. The nonconformance control practice includes the following elements:

- o Establish Disposition
- o Documentation and Reporting
- o Review, Evaluation and Disposition

All reports of deviations that are proposed for disposition in such a way that the finished item or completed service will not conform to the approved requirements are processed for approval in accordance with procedures established to provide a level of approval equivalent to the original approval of the requirements that will not be met as a result of the proposed disposition. These reports are part of the documentation required at the plant site.

Errors or deficiencies found in design and construction reported to or discovered by the Owner which could adversely affect safety-related structures, systems or components and which represent a breakdown in the quality assurance program, deficiency in final design, deficiency in construction or deviation from performance specifications are evaluated by Engineering, Public Safety and Quality Assurance for consideration as a reportable deficiency under 10 CFR 50.55(E). If it is concluded that the error or deficiency comes under this paragraph, the deficiency together with the proposed corrective action is reported to the Nuclear Regulatory Commission according to regulations. Defects or noncompliance in the plant or basic component supplied to the plant that are reported to, or discovered by the Owner which could adversely affect safety related functions of the plant are evaluated by Engineering, Public Safety, and Quality Assurance for consideration as a reportable deficiency under 10 CFR 21. If it is concluded that the defects or noncompliance is reportable under part 21, the deficiency is reported to the Nuclear Regulatory Commission according to regulations. The deficiency, whether reportable or not, is further evaluated against the procedural requirements that should have prevented the occurrence. When the procedural system is deficient, the affected organization is required to take whatever steps are necessary to achieve appropriate corrective action to the system to preclude recurrence of the deficiency. The deficiency is reported within the Project via an unusual occurrence report as described in Section 16.

The Owner Quality Assurance organization also participates in and monitors the execution of the nonconformance control practices and periodically audits or arranges for an independent audit of the control practices to assure implementation and adequacy.

15.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each participant, who is assigned responsibility for procurement, manufacturing, or construction of items of the CRBRP, is required by contract to establish and implement a practice for the control of nonconforming materials, parts or components. These nonconformance control practices will include the following elements:

- The identification, documentation, segregation where practicable, review, disposition, and notification of affected organizations of nonconforming materials, parts, components, or services (including computer codes) is undertaken if disposition is other than to scrap.

- Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconformance, and the inspection requirements; and includes signature approval of the disposition. Nonconformances are corrected or resolved prior to the initiation of the preoperational test program on the item.

- Provisions are established identifying those individuals or groups delegated the responsibility and authority to approve the dispositioning and closeout of nonconforming items.

- Nonconforming items are segregated, where practicable, from acceptable items and identified as discrepant until properly dispositioned.

- Acceptability of rework or repair of materials, parts, components, systems, and structures is verified by reinspecting the items as originally inspected or by a method which is at least equal to the original inspection method; inspection, rework, and repair procedures are documented.

- Nonconformance reports dispositioned "use as is" or "use as repaired" or "use as modified" are made part of the inspection records and forwarded with the hardware to the Owner.

- Nonconformance reports are periodically analyzed to show quality trends, and the significant results are forwarded to management for review and assessment.

These practices will assure that nonconforming items are reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures. They will include measures which control further processing, delivery or installation pending proper disposition of the deficiency.

The Owner monitors the major participants nonconformance control practices and periodically audits the major participant's nonconformance control practices to assure implementation and adequacy.

16.0 CORRECTIVE ACTION

16.1 OWNER IMPLEMENTATION

The Owner has established and implemented a system for corrective action wherein conditions adverse to quality such as failures, nonconformances, malfunctions, deficiencies, deviations and defective material and equipment that are required for reliable and safe operation of the plant are reported to the Owner through nonconformance and unusual occurrence reporting procedures. Quality assurance activities found deficient by Owner reviews and audits of the participants are also reported. The corrective action system includes the following elements:

- Evaluation of nonconformances and determination of the need for corrective action in accordance with established procedures.

- Prompt corrective action initiated and documented following the determination of a nonconformance to preclude the recurrence of those adverse conditions significant to quality.

- Follow-up reviews conducted by the quality assurance organization to verify proper implementation of corrective actions and to close out the corrective action documentation in a timely manner.

- Adverse conditions significant to quality, the cause of the conditions, and the corrective action taken are documented and reported to appropriate levels of management for review and assessment.

The Owner Quality Assurance organization executes the corrective action system and arranges for periodic independent audit of the system to assure implementation and adequacy.

16.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each participant is required by contract to establish and implement a corrective action system. Each system will include the following elements:

- o Evaluation of nonconformances and determination of the need for corrective action in accordance with established procedures.
- o Prompt corrective action initiated and documented following the determination of a nonconformance to preclude the recurrence of those adverse conditions significant to quality.
- o Follow-up reviews conducted by the quality assurance organization to verify proper implementation of corrective actions and to close out the corrective action documentation in a timely manner.
- o Adverse conditions significant to quality, the cause of the conditions, and the corrective action taken are documented and reported to appropriate levels of management for review and assessment.

The Owner monitors major participant corrective action systems and periodically audits the participant's system to assure implementation and adequacy.

17.0 QUALITY ASSURANCE RECORDS

17.1 OWNER IMPLEMENTATION

The Owner has established a quality assurance records system that provides for the collection, storage and maintenance of Owner prepared records in accordance with approved records management procedures.

The Owner has delegated execution responsibility for other records preparation and initial collection, storage and maintenance to the other major participants by contract. This includes those official records directly related to structures, systems and components of the plant that are prepared by and used in design, procurement, manufacturing, construction and operation. In progressive stages as required by code, standard, regulation or specification, these records will be turned over to the Owner.

The Owner Quality Assurance records system provides for overall coordination of records management practices as well as the collection, storage and maintenance of those records resulting from the Owner activities and those records from major participants, as they are turned over to the Owner. Included are those records that are necessary to define the overall quality of the plant and provide objective evidence of quality achievement. The system includes provisions that ensure:

- o Records are maintained to provide documentary evidence of the quality of items and the activities affecting quality.
- o QA records include: operating logs; results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; calibration procedures and reports, nonconforming and corrective action reports, and other associated documents.
- o Records are readily identifiable and retrievable.
- o Requirements and organizational responsibilities for record transmittals, retention, and maintenance subsequent to completion of work are consistent with applicable codes, standards, and procurement documents.
- o Inspection and test records contain the following:

    a. A description of the type of observation.
    b. Evidence of completing and verifying a manufacturing, inspection, or test operation.
    c. The date and results of the inspection or test.
    d. Information related to nonconformances and other conditions adverse to quality.
    e. Inspector or data recorder identification.
    f. A statement as to the acceptability of the results.
    g. Action taken to resolve any discrepancies noted.

- o Record storage facilities are constructed, located, and secured to prevent destruction of the records by fire, flooding, theft and deterioration by environmental conditions, such as temperature or humidity.

The Owner Quality Assurance organization participates in and monitors the implementation of the records system. Periodically the Quality Assurance organization audits or arranges for independent audit of the records system to assure implementation and adequacy.

17.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each major Project participant is required by contract to maintain a quality assurance records system. Each records system will contain provisions that ensure:

- o Records are maintained to provide documentary evidence of the quality of items and the activities affecting quality.
- o QA records include: operating logs; results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports, and nonconforming and corrective action reports.

- Records are readily identifiable and retrievable.

- Requirements and organizational responsibilities for record transmittals, retention, and maintenance subsequent to completion of work are consistent with applicable codes, standards, and procurement documents.

- Inspection and test records contain the following:

  a. A description of the type of observation.
  b. Evidence of completing and verifying a manufacturing, inspection or test operation.
  c. The date and results of the inspection or test.
  d. Information related to nonconformances and other conditions adverse to quality.
  e. Inspector or data recorder identification.
  f. A statement as to the acceptability of the results.
  g. Action taken to resolve any discrepancies noted.

- Record storage facilities are constructed, located and secured to prevent destruction of the records by fire, flooding, theft, and deterioration by environmental conditions such as temperature or humidity.

The Owner monitors major participant quality assurance records systems and periodically audits the participants' systems to assure implementation and adequacy.

18.0 AUDITS

18.1 OWNER IMPLEMENTATION

The Owner has established and implemented a quality assurance audit practice which is used to provide a comprehensive independent verification and evaluation, both internally and externally, of the status and adequacy of the overall Project quality assurance program methods, quality-related procedures and activities. This practice includes the Owner program as well as the programs of the other major program participants and their suppliers. This practice is also designed to assure that procedures and activities are meaningful and comply with the overall Project Quality Assurance Program requirements. The practice includes the following elements:

- Planning

- Evaluation of Quality Assurance Methods

- Activity Audits

- Product Audits

- Record Audits

- Reporting and Corrective Action

The Quality Assurance audit practice was established during the project planning and conceptual design stage with the scope and frequency of the audits planned and scheduled based upon the reliable and safe operation of the plant and was initiated early enough to assure effective quality assurance practices during design, procurement, manufacturing, construction and installation, inspection and test. Audits are planned in a general way on an annual basis with a more detailed plan and schedule prepared and issued on a quarterly basis. Audits are planned to cover not only the evaluation of internal practices of the Owner's Program but also to cover the practices of each of the major program participants at the second level of the program who have a direct interface with the Owner. The audit plan for each major participant program is designed to include an objective evaluation of quality-related practices, procedures and instructions; the effectiveness of implementation; and the conformance with policy directives. These audits include the evaluation of work areas, activities including personnel training and indoctrination, processes, and items; and the review of documents and records to ensure that the quality assurance programs are effective and properly implemented. In each major participant program, identified elements of interface control are evaluated with respect to each participant's internal activities as well as interfacing activities with his customer and his subcontractors. The audit plan is supplemented by unscheduled audits where the need becomes evident.

Each audit is conducted to pre-established written procedures and check lists that include a detailed plan for the audit with a prepared check list of items to be investigated; a meeting with responsible management personnel before the audit to review scope, purpose, and schedule of the audit and at the conclusion, to review audit findings with management having responsibility in the area audited. The need for any corrective actions is established and the audit results are documented in a formal report. Each audit is conducted by trained personnel that do not have direct responsibilities in the areas being audited.

Responsible management is then required to take the necessary action to correct the deficiencies revealed by the audit and to provide the auditing organization with a statement of proposed and completed corrective action taken. Deficient areas are monitored and promptly re-audited, when necessary, until corrections have been accomplished.

Audit summaries are provided in both internal and external monthly progress and status reports. The audit reports and data are analyzed by the quality assurance organization for quality trends indicating any quality problems and the effectiveness of the quality assurance program, including the need for reaudit of deficient areas. These reports are provided to management for review and assessment.

The Owner audit practice is performed as a minimum in those areas of the Overall Project Quality Assurance Program where the requirements of 10 CFR 50, Appendix B are being implemented. Each major Project participant has prepared a matrix showing which procedures are used for implementation of each of the eighteen criteria of Appendix B. The activities and practices which carry out these procedures are audited on a pre-scheduled basis. These activities include:

1. The preparation, review, approval, and control of the PSAR, designs, specifications, procurement documents, instructions, procedures and drawings.

2. The determination of site features which affect plant safety (e.g., core sampling, site preparation, and meteorology).

3. Request for proposals and evaluations of bids.

4. Indoctrination and training programs.

5. Interface control among the applicant and the principal contractors.

6. Calibration and nonconformance control systems.

7. SAR and SSAR commitments.

8. Activities associated with computer codes.

18.2 REQUIREMENTS OF OTHER PARTICIPANTS

Each major Project participant is required by contract to establish and implement an audit practice that satisfies the quality assurance requirements. Each participant's audit practice will include the following elements.

- o Planning
- o Evaluation of Quality Assurance Methods
- o Activity Audits
- o Product Audits
- o Nondestructive Examination Audits
- o Records Audits
- o Reporting and Corrective Action

A periodic summary of the actions and accomplishments of each major participant's audit practices is made to the Owner for review and evaluation. The Owner monitors major participant audit practices and periodically audits the participant's practices to assure implementation and adequacy.

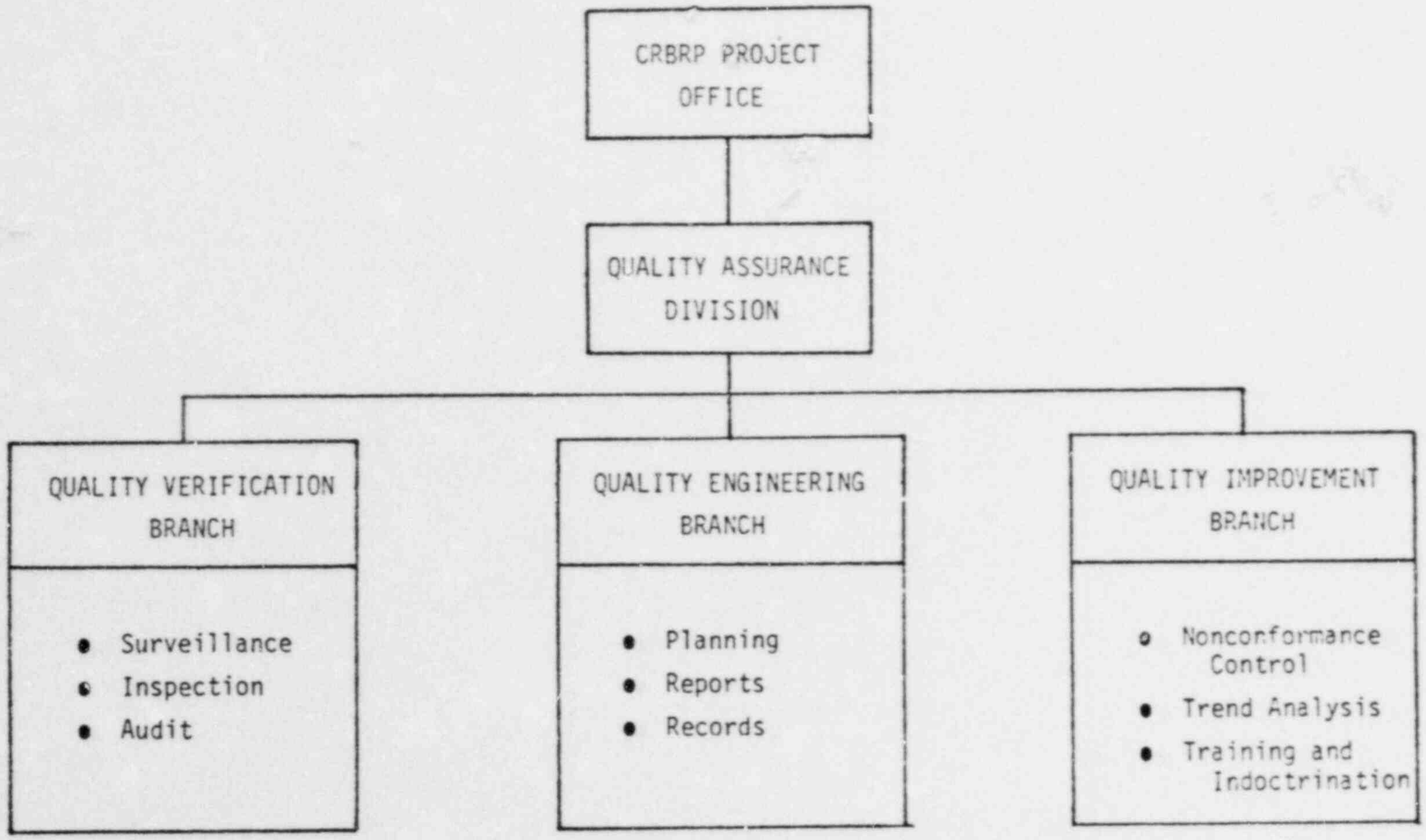

Figure 17A-1. ORGANIZATION OF QUALITY ASSURANCE DIVISION

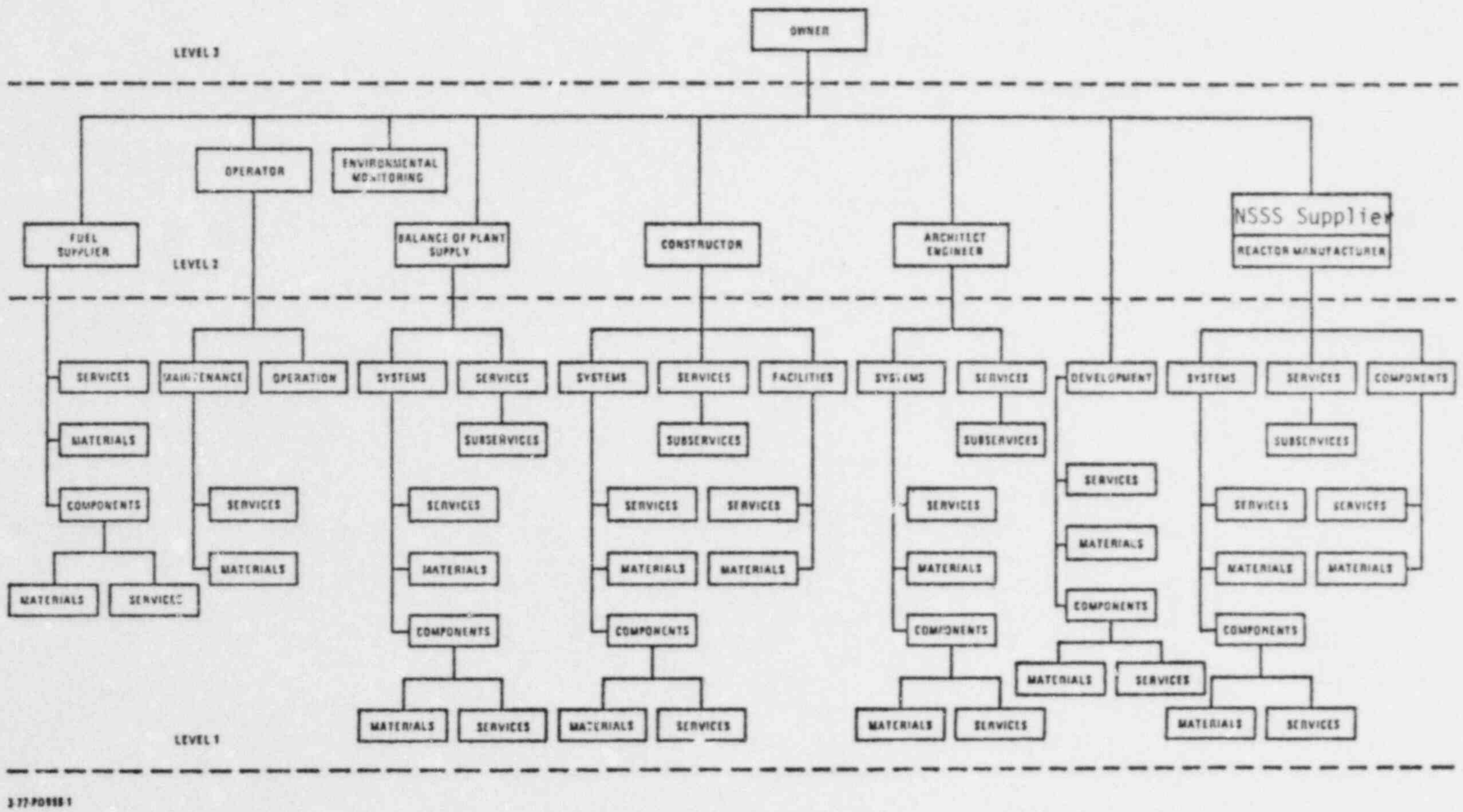

Figure 17A-2. CRBRP Quality Assurance Program Functional Organization of Program Responsibility

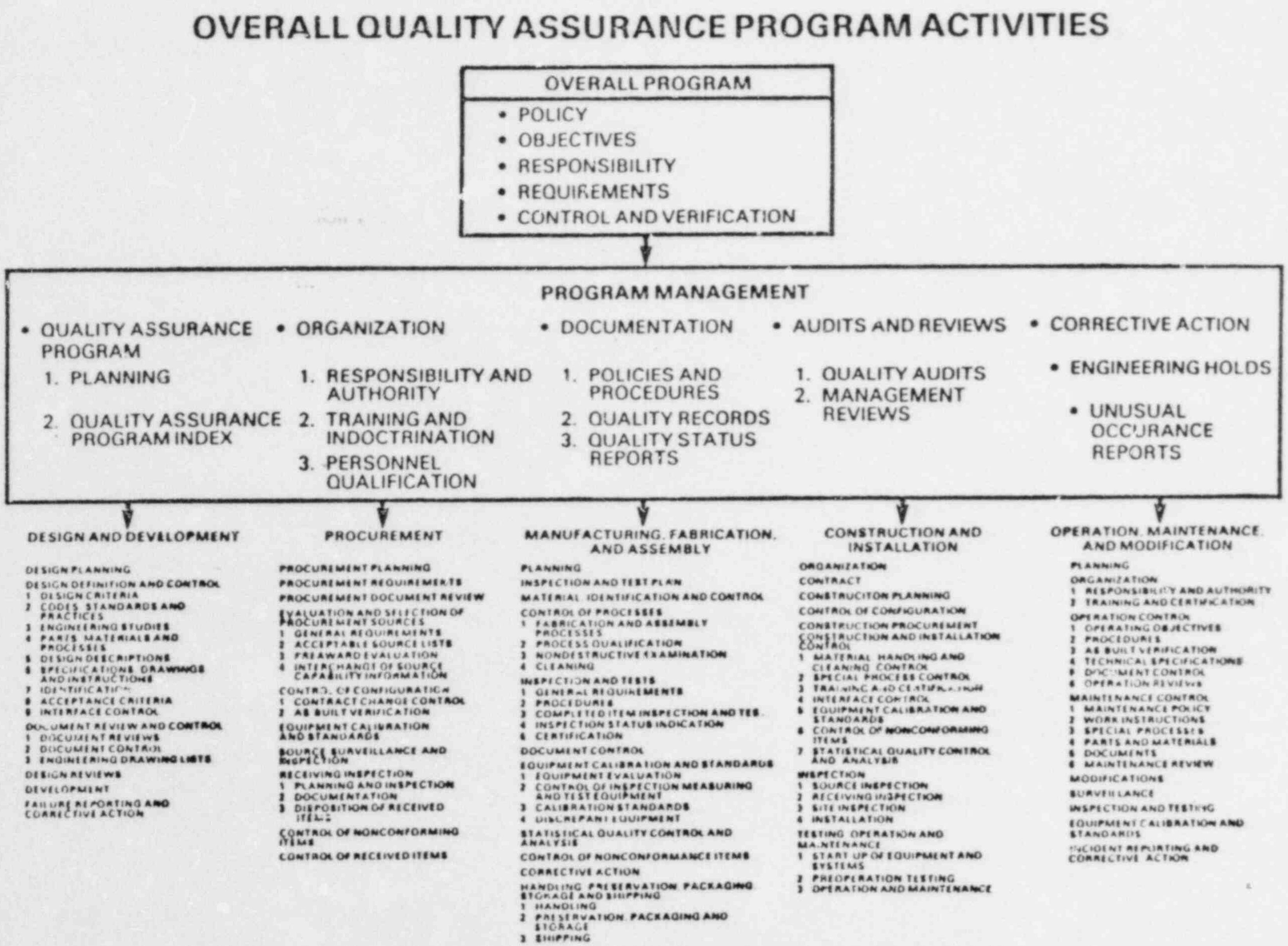

FIGURE 17A-3. MAJOR ELEMENTS OF OVERALL QUALITY ASSURANCE PROGRAM

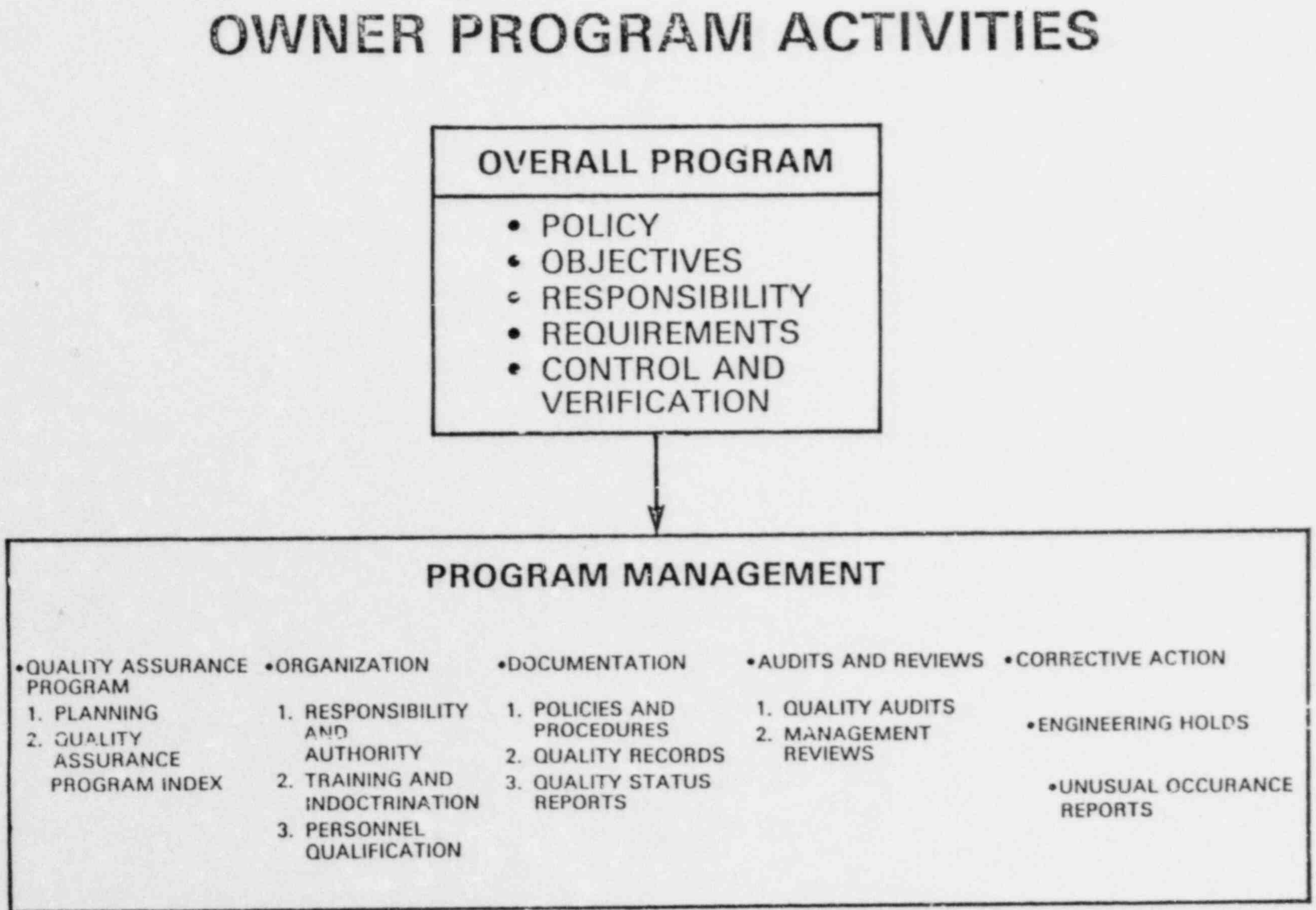

FIGURE 17A-4. MAJOR ELEMENTS OF THE OWNER PROGRAM

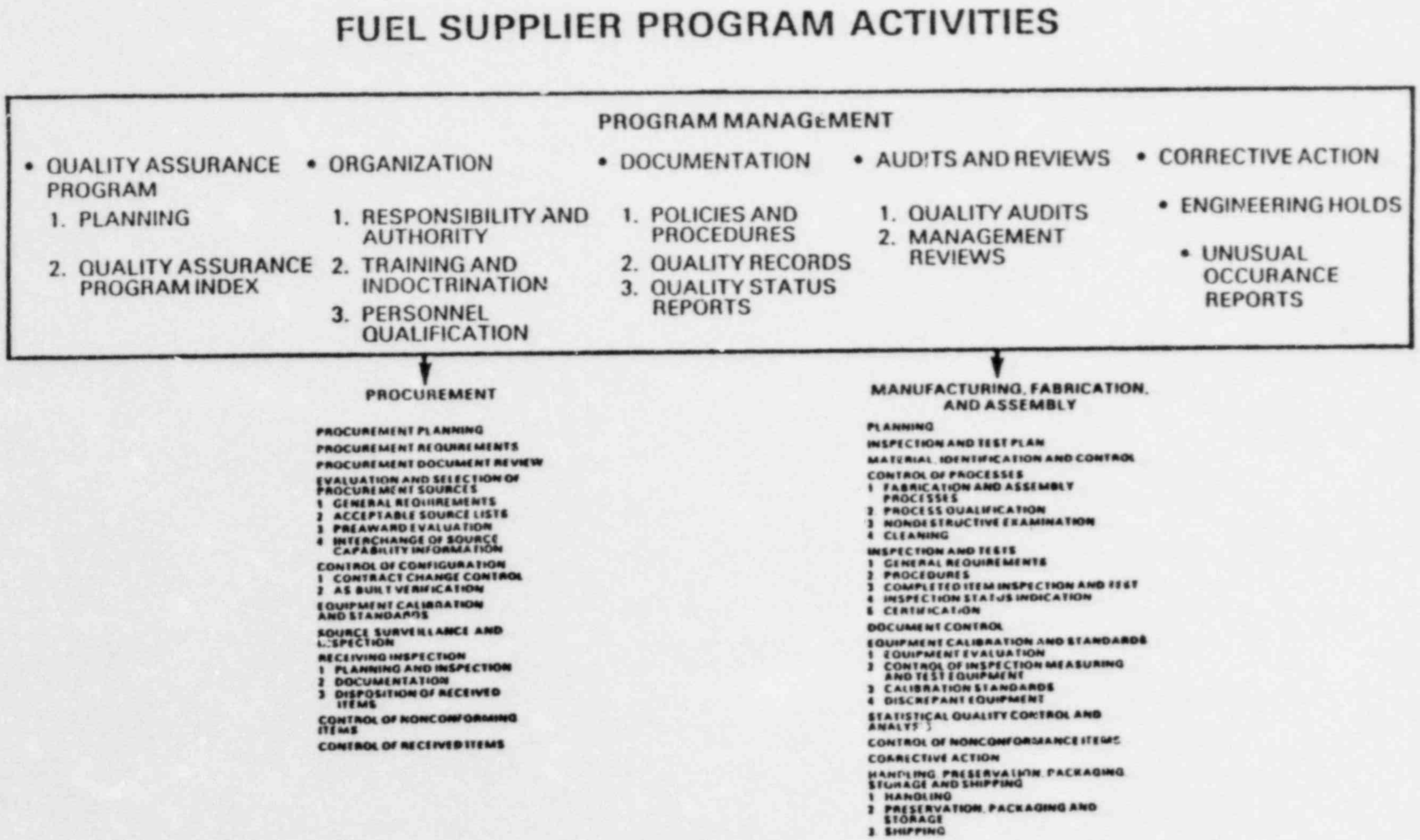

FIGURE 17A-5. MAJOR ELEMENTS OF THE FUEL SUPPLIER PROGRAM

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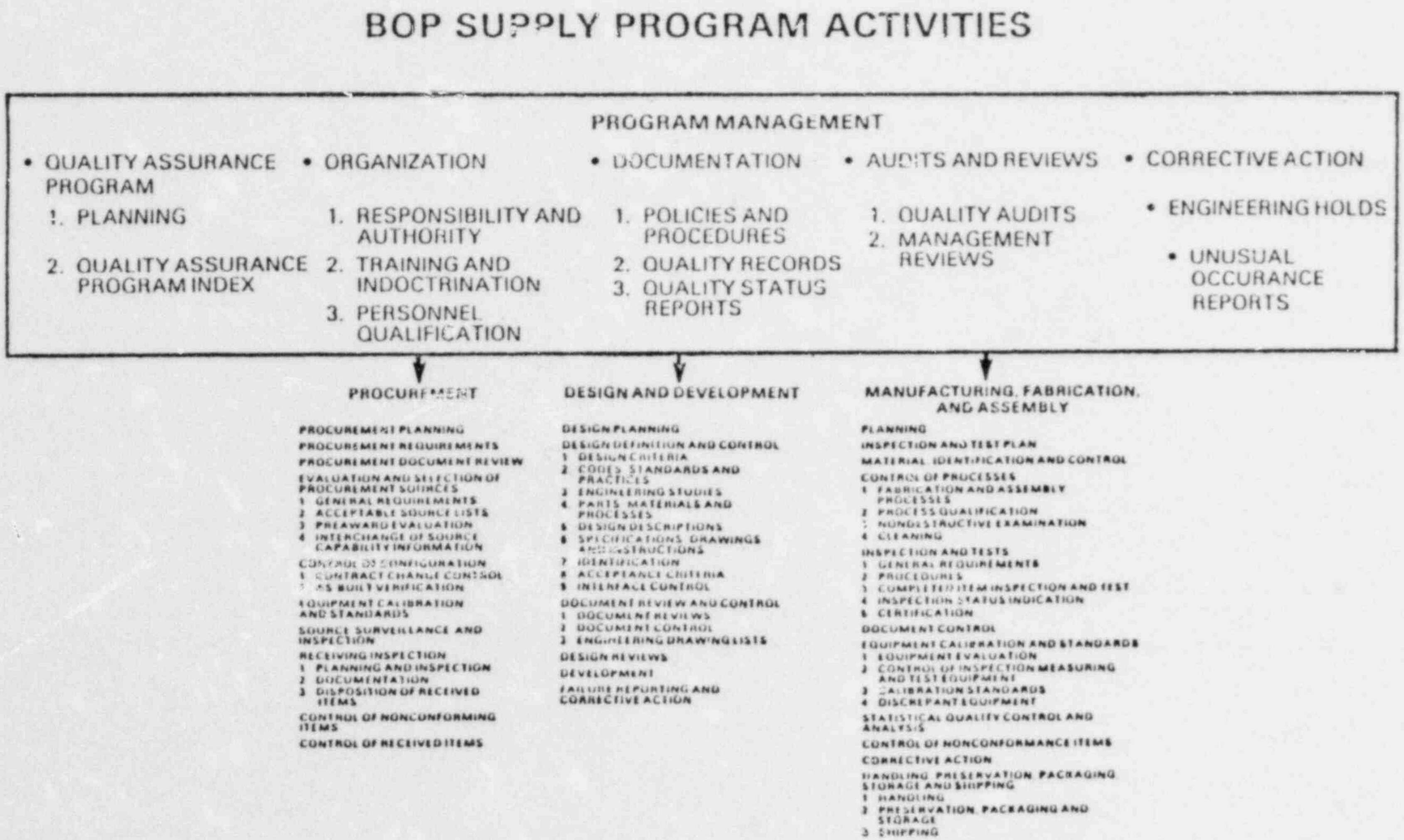

FIGURE 17A-6. MAJOR ELEMENTS OF THE BOP SUPPLY PROGRAM

NSSS/RM PROGRAM ACTIVITIES

PROGRAM MANAGEMENT

- QUALITY ASSURANCE PROGRAM
  1. PLANNING
  2. QUALITY ASSURANCE PROGRAM INDEX
- ORGANIZATION
  1. RESPONSIBILITY AND AUTHORITY
  2. TRAINING AND INDOCTRINATION
  3. PERSONNEL QUALIFICATION
- DOCUMENTATION
  1. POLICIES AND PROCEDURES
  2. QUALITY RECORDS
  3. QUALITY STATUS REPORTS
- AUDITS AND REVIEWS
  1. QUALITY AUDITS
  2. MANAGEMENT REVIEWS
- CORRECTIVE ACTION
  - ENGINEERING HOLDS
  - UNUSUAL OCCURANCE REPORTS

DESIGN AND DEVELOPMENT

DESIGN PLANNING

DESIGN DEFINITION AND CONTROL
1. DESIGN CRITERIA
2. CODES STANDARDS AND PRACTICES
3. ENGINEERING STUDIES
4. PARTS MATERIALS AND PROCESSES
5. DESIGN DESCRIPTIONS
6. SPECIFICATIONS DRAWINGS AND INSTRUCTIONS
7. IDENTIFICATION
8. ACCEPTANCE CRITERIA
9. INTERFACE CONTROL

DOCUMENT REVIEW AND CONTROL
1. DOCUMENT REVIEWS
2. DOCUMENT CONTROL
3. ENGINEERING DRAWING LISTS

DESIGN REVIEWS

DEVELOPMENT

FAILURE REPORTING AND CORRECTIVE ACTION

PROCUREMENT

PROCUREMENT PLANNING

PROCUREMENT REQUIREMENTS

PROCUREMENT DOCUMENT REVIEW

EVALUATION AND SELECTION OF PROCUREMENT SOURCES
1. GENERAL REQUIREMENTS
2. ACCEPTABLE SOURCE LISTS
3. PREAWARD EVALUATION
4. INTERCHANGE OF SOURCE CAPABILITY INFORMATION

CONTROL OF CONFIGURATION
1. CONTRACT CHANGE CONTROL
2. AS BUILT VERIFICATION

EQUIPMENT CALIBRATION AND STANDARDS

SOURCE SURVEILLANCE AND INSPECTION

RECEIVING INSPECTION
1. PLANNING AND INSPECTION
2. DOCUMENTATION
3. DISPOSITION OF RECEIVED ITEMS

CONTROL OF NONCONFORMING ITEMS

CONTROL OF RECEIVED ITEMS

MANUFACTURING, FABRICATION, AND ASSEMBLY

PLANNING

INSPECTION AND TEST PLAN

MATERIAL IDENTIFICATION AND CONTROL

CONTROL OF PROCESSES
1. FABRICATION AND ASSEMBLY PROCESSES
2. PROCESS QUALIFICATION
3. NONDESTRUCTIVE EXAMINATION
4. CLEANING

INSPECTION AND TESTS
1. GENERAL REQUIREMENTS
2. PROCEDURES
3. COMPLETED ITEM INSPECTION AND TEST
4. INSPECTION STATUS INDICATION
5. CERTIFICATION

DOCUMENT CONTROL

EQUIPMENT CALIBRATION AND STANDARDS
1. EQUIPMENT EVALUATION
2. CONTROL OF INSPECTION MEASURING AND TEST EQUIPMENT
3. CALIBRATION STANDARDS
4. DISCREPANT EQUIPMENT

STATISTICAL QUALITY CONTROL AND ANALYSIS

CONTROL OF NONCONFORMANCE ITEMS

CORRECTIVE ACTION

HANDLING PRESERVATION PACKAGING STORAGE AND SHIPPING
1. HANDLING
2. PRESERVATION PACKAGING AND STORAGE
3. SHIPPING

FIGURE 17A-7. MAJOR ELEMENTS OF THE NSSS/RM PROGRAM

A-E PROGRAM ACTIVITIES

PROGRAM MANAGEMENT

- QUALITY ASSURANCE PROGRAM
  1. PLANNING
  2. QUALITY ASSURANCE PROGRAM INDEX
- ORGANIZATION
  1. RESPONSIBILITY AND AUTHORITY
  2. TRAINING AND INDOCTRINATION
  3. PERSONNEL QUALIFICATION
- DOCUMENTATION
  1. POLICIES AND PROCEDURES
  2. QUALITY RECORDS
  3. QUALITY STATUS REPORTS
- AUDITS AND REVIEWS
  1. QUALITY AUDITS
  2. MANAGEMENT REVIEWS
- CORRECTIVE ACTION
  - ENGINEERING HOLDS
  - UNUSUAL OCCURANCE REPORTS

DESIGN AND DEVELOPMENT

DESIGN PLANNING

DESIGN DEFINITION AND CONTROL
1. DESIGN CRITERIA
2. CODES, STANDARDS AND PRACTICES
3. ENGINEERING STUDIES
4. PARTS, MATERIALS AND PROCESSES
5. DESIGN DESCRIPTIONS
6. SPECIFICATIONS, DRAWINGS AND INSTRUCTIONS
7. IDENTIFICATION
8. ACCEPTANCE CRITERIA
9. INTERFACE CONTROL

DOCUMENT REVIEW AND CONTROL
1. DOCUMENT REVIEWS
2. DOCUMENT CONTROL
3. ENGINEERING DRAWING LISTS

DESIGN REVIEWS

DEVELOPMENT

FAILURE REPORTING AND CORRECTIVE ACTION

PROCUREMENT

PROCUREMENT PLANNING

PROCUREMENT REQUIREMENTS

PROCUREMENT DOCUMENT REVIEW

EVALUATION AND SELECTION OF PROCUREMENT SOURCES
1. GENERAL REQUIREMENTS
2. ACCEPTABLE SOURCE LISTS
3. PREAWARD EVALUATION
4. INTERCHANGE OF SOURCE CAPABILITY INFORMATION

CONTROL OF CONFIGURATION
1. CONTRACT CHANGE CONTROL
2. AS BUILT VERIFICATION

EQUIPMENT CALIBRATION AND STANDARDS

SOURCE SURVEILLANCE AND INSPECTION

RECEIVING INSPECTION
1. PLANNING AND INSPECTION
2. DOCUMENTATION
3. DISPOSITION OF RECEIVED ITEMS

CONTROL OF NONCONFORMING ITEMS

CONTROL OF RECEIVED ITEMS

MANUFACTURING, FABRICATION, AND ASSEMBLY

PLANNING

INSPECTION AND TEST PLAN

MATERIAL IDENTIFICATION AND CONTROL

CONTROL OF PROCESSES
1. FABRICATION AND ASSEMBLY PROCESSES
2. PROCESS QUALIFICATION
3. NONDESTRUCTIVE EXAMINATION
4. CLEANING

INSPECTION AND TESTS
1. GENERAL REQUIREMENTS
2. PROCEDURES
3. COMPLETED ITEM INSPECTION AND TEST
4. INSPECTION STATUS INDICATION
5. CERTIFICATION

DOCUMENT CONTROL

EQUIPMENT CALIBRATION AND STANDARDS
1. EQUIPMENT EVALUATION
2. CONTROL OF INSPECTION MEASURING AND TEST EQUIPMENT
3. CALIBRATION STANDARDS
4. DISCREPANT EQUIPMENT

STATISTICAL QUALITY CONTROL AND ANALYSIS

CONTROL OF NONCONFORMANCE ITEMS

CORRECTIVE ACTION

HANDLING, PRESERVATION, PACKAGING, STORAGE AND SHIPPING
1. HANDLING
2. PRESERVATION, PACKAGING AND STORAGE
3. SHIPPING

FIGURE 17A-8 MAJOR ELEMENTS OF THE A-E PROGRAM

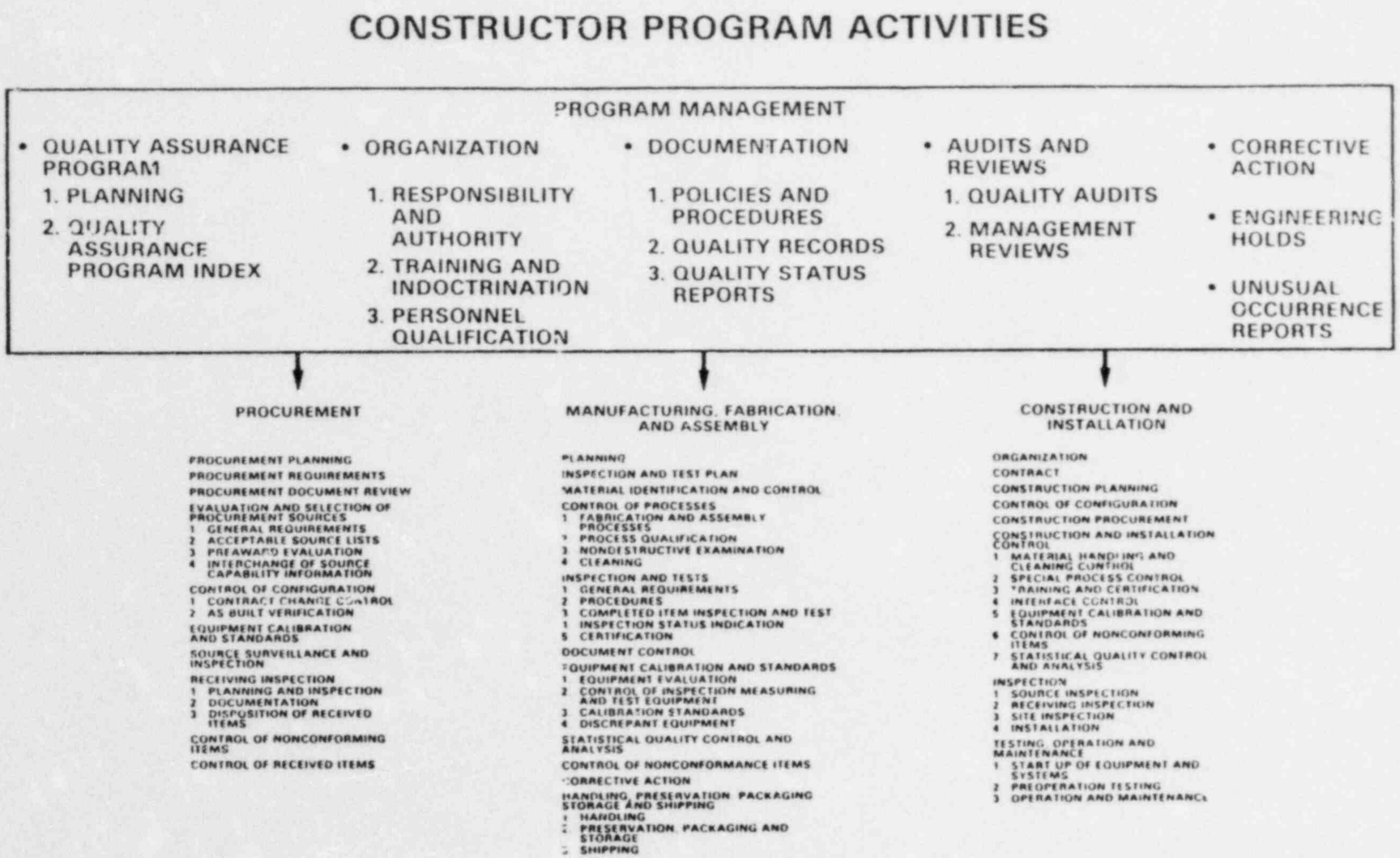

FIGURE 17A-9 MAJOR ELEMENTS OF THE CONSTRUCTOR PROGRAM

OPERATOR PROGRAM ACITIVITES

PROGRAM MANAGEMENT

- QUALITY ASSURANCE PROGRAM
  1. PLANNING
  2. QUALITY ASSURANCE PROGRAM INDEX
- ORGANIZATION
  1. RESPONSIBILITY AND AUTHORITY
  2. TRAINING AND INDOCTRINATION
  3. PERSONNEL QUALIFICATION
- DOCUMENTATION
  1. POLICIES AND PROCEDURES
  2. QUALITY RECORDS
  3. QUALITY STATUS REPORTS
- AUDITS AND REVIEWS
  1. QUALITY AUDITS
  2. MANAGEMENT REVIEWS
- CORRECTIVE ACTION
  - ENGINEERING HOLDS
  - UNUSUAL OCCURANCE REPORTS

OPERATION, MAINTENANCE, AND MODIFICATION

PLANNING

ORGANIZATION
1 RESPONSIBILITY AND AUTHORITY
2 TRAINING AND CERTIFICATION

OPERATION CONTROL
1 OPERATING OBJECTIVES
2 PROCEDURES
3 AS BUILT VERIFICATION
4 TECHNICAL SPECIFICATIONS
5 DOCUMENT CONTROL
6 OPERATION REVIEWS

MAINTENANCE CONTROL
1 MAINTENANCE POLICY
2 WORK INSTRUCTIONS
3 SPECIAL PROCESSES
4 PARTS AND MATERIALS
5 DOCUMENTS
6 MAINTENANCE REVIEW

MODIFICATIONS

SURVEILLANCE

INSPECTION AND TESTING

EQUIPMENT CALIBRATION AND STANDARDS

INCIDENT REPORTING AND CORRECTIVE ACTION

FIGURE 17A-10 MAJOR ELEMENTS OF THE OPERATOR PROGRAM

EM PROGRAM ACTIVITIES

PROGRAM MANAGEMENT

- QUALITY ASSURANCE PROGRAM
  1. PLANNING
  2. QUALITY ASSURANCE PROGRAM INDEX
- ORGANIZATION
  1. RESPONSIBILITY AND AUTHORITY
  2. TRAINING AND INDOCTRINATION
  3. PERSONNEL QUALIFICATION
- DOCUMENTATION
  1. POLICIES AND PROCEDURES
  2. QUALITY RECORDS
  3. QUALITY STATUS REPORTS
- AUDITS AND REVIEWS
  1. QUALITY AUDITS
  2. MANAGEMENT REVIEWS
- CORRECTIVE ACTION
  - ENGINEERING HOLDS
  - UNUSUAL OCCURANCE REPORTS

DESIGN AND DEVELOPMENT

DESIGN PLANNING

DESIGN DEFINITION AND CONTROL
1. ENGINEERING STUDIES
2. SPECIFICATIONS, DRAWINGS AND INSTRUCTIONS

DOCUMENT REVIEW AND CONTROL
1. DOCUMENT REVIEWS
2. DOCUMENT CONTROL
3. ENGINEERING DRAWING LISTS

DESIGN REVIEWS

PROCUREMENT

PROCUREMENT PLANNING

PROCUREMENT REQUIREMENTS

PROCUREMENT DOCUMENT REVIEW

EVALUATION AND SELECTION OF PROCUREMENT SOURCES
1. GENERAL REQUIREMENTS
2. ACCEPTABLE SOURCE LISTS
3. PREAWARD EVALUATION
4. INTERCHANGE OF SOURCE CAPABILITY INFORMATION

MANUFACTURING, FABRICATION, AND ASSEMBLY

INSPECTION AND TESTS
1. PROCEDURES

FIGURE 17A-11 MAJOR ELEMENTS OF THE EM PROGRAM

REQUIREMENT OF RDT F2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
2.	Management & Planning			
2.2	Quality Assurance Program		Quality Assurance Program Description Quality Assurance Charter	
2.2.1	Planning	 CRP-EN-09 CRP-PC-02 CRP-PC-05 CRP-QA-02	Quality Assurance Program Description Preparation and Maintenance of the Project Level 1 Schedule Preparation and Maintenance of the CRBRP Project Level 0 Schedule Preparation and Maintenance of the Work Breakdown Structure (WBS) Activity Planning	
2.3	Organization			
2.3.1	Responsibility and Authority	 CRP-DR-02	Quality Assurance Program Description Quality Assurance Charter All Procedures Organization Plan and Functional Statements	Responsibility Sections
2.3.2	Training and Indoctrination	CRP-QA-24	Personnel Indoctrination	
2.3.3	Personnel Qualification	CRP-QA-25 CRP-QA-26	Administration of Personnel Certification and Records Personnel Certification	
2.4	Documentation			
2.4.1	Policies and Procedures	 CRP-AA-01 CRP-AA-03 CRP-AA-11 CRP-AA-14 CRP-QA-20	Quality Assurance Program Description All Procedures CRBRP Management Policies and Requirements Management Procedures Preparation of Correspondence Control of Project Office Procedures Manual Controlled Documents Preparation, Maintenance and Control of Project Office Quality Assurance Manual	Policy Sections

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2

Figure 17A-12 OWNER QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
2.4.2	Quality Records	CRP-AA-02 CRP-QA-07 CRP-QA-23	Filing Procedure for Official Project Files Quality Records Preparation, Transfer, and Receipt of Project Office Quality Records	
2.4.3	Quality Status Reports	CRP-AA-07 CRP-PC-03 CRP-QA-01 CRP-QA-08	Reports Control Program CRBRP Project Monthly Progress Report Quality Assurance Program Management Review Meetings Quality Assurance Program Progress and Status Review and Reporting	
2.5	Audits and Reviews			
2.5.1	Quality Audits			See Section 8
2.5.2	Management Reviews	CRP-QA-01	Quality Assurance Program Management Review Meetings	
2.6	Corrective Action	CRP-AA-06 CRP-QA-04 CRP-QA-05 CRP-QA-06 CRP-QA-09 CRP-QA-27	Centralized Action Correspondence Control System Corrective Action Requests Processing of Unusual Occurrence Reports Nonconformance, Unusual Occurrence and Corrective Action Analysis Quality Trend Analysis Unusual Occurrence Report Preparation and Disposition	
2.7	Engineering Holds	CRP-EN-01 CRP-EN-02	Design Control Processing Principal Design Documents	Partially Delegated
2.8	Unusual Occurrence Reporting	CRP-QA-05 CRP-QA-27	Processing of Unusual Occurrence Reports Unusual Occurrence Report Preparation and Disposition	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2

Figure 17A-12 (Cont'd.). Owner Quality Assurance Program Index

REQUIREMENT OF RDT F2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
3.	Design and Development			Partially Delegated
3.2	Design Planning	CRP-AA-07	Reports Control Program	
		CRP-EN-09	Preparation and Maintenance of the Project Level 1 Schedule	
		CRP-PC-03	CRBRP Project Monthly Progress Report	
		CRP-PC-05	Preparation and Maintenance of the Work Breakdown Structure (WBS)	
3.3	Design Definition and Control	CRP-DR-01	Changes in Project Scope or Major Deviations from The Reference Design	
		CRP-EN-01	Design Control	
		CRP-PC-05	Preparation and Maintenance of the Work Breakdown Structure (WBS)	
3.4	Document Review and Control	CRP-AA-03	Preparation of Correspondence	
		CRP-AA-04	Incoming Mail	
		CRP-AA-07	Reports Control Program	
		CRP-AA-11	Control of Project Office Procedures Manual	
		CRP-EN-11	Review, Approval, and Issuance of Construction Documentation Requiring Project Office Approval	
		CRP-AA-14	Controlled Documents	
		CRP-CN-01	Processing of Construction Data Submitted by the Constructor	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-EN-04	Processing Engineering Changes	
		CRP-EN-05	Configuration Control Board Actions	
		CRP-EN-07	Technical Control of CRBRP Test Programs	
		CRP-EN-10	Design Document Control	
		CRP-OP-02	Operations Division Review and Concurrence with Engineering Design Data	
		CRP-OP-03	Operations Division Review and Concurrence with Licensing Data	
		CRP-PC-06	Revision to the CRBRP Project Management Policies and Requirements	
		CRP-PS-01	Preparation of Amendments to and Maintenance of the PSAR and ER	
		CRP-PS-02	Preparation of Responses to Nuclear Regulatory Commission Questions	

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FIGURE 17A-12 (Cont'd) OWNER QUALITY ASSURANCE PROGRAM INDEX

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REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
		CRP-PS-06	Distribution, Evaluation, and Preparation of Responses to NRC's Office of Inspection and Enforcement Bulletins, Circulars, and Notices	
		CRP-PS-03	Preparation and Approval of Responses to Request for Licensing and Safety Information	
		CRP-PS-04	Preparation and Compliance with Non-NRC Permits, Licenses, and Approvals	
		CRP-PS-05	Maintenance of Consistency Between the PSAR and the Project Baseline Documentation	
		CRP-QA-10	Quality Assurance Review and Approval of Engineering Documents	
		CRP-QA-11	Quality Assurance Review of Procurement Documents	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
3.5	Design Review	CRP-EN-03	Design Reviews	
3.6	Development	CRP-EN-06	Development Program Technical Management	
		CRP-PC-05	Preparation and Maintenance of the Work Breakdown Structure (WBS)	
		CRP-QA-10	Quality Assurance Review and Approval of Engineering Documents	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F2-2

FIGURE 17A-12 (CONT'D.) OWNER QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
4.	Procurement			Partially Delegated
4.2	Procurement Planning	CRP-CN-09 CRP-PC-03 CRP-PC-05 CRP-QA-02	Preparation and Maintenance of the Project Level 1 Schedule CRBRP Project Monthly Progress Report Preparation and Maintenance of the Work Breakdown Structure (WBS) Activity Planning	
4.3	Procurement Requirements	CRP-CN-01 CRP-EN-02 CRP-EN-04 CRP-EN-05 CRP-OC-01 CRP-OP-02 CRP-PR-03 CRP-QA-10	Processing of Construction Data Submitted by the Constructor Processing Principal Design Documents Processing Engineering Changes Configuration Control Board Actions **Control of Modifications to Principal Project Agreements** Operations Division Review and Concurrence with Engineering Design Data Contract and Subcontract Review Actions Quality Assurance Review and Approval of Engineering Documents	
4.4	Procurement Document Review	CRP-CN-01 CRP-EN-02 CRP-EN-04 CRP-EN-05 CRP-PR-03 CRP-QA-11	Processing of Construction Data Submitted by the Constructor Processing Principal Design Documents Processing Engineering Changes Configuration Control Board Actions Contract and Subcontract Review Actions Quality Assurance Review of Procurement Documents	
4.5	Evaluation and Selection of Procurement Sources	CRP-PR-03	Contract and Subcontract Review Actions	

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FIGURE 17A-12 (CONT'D.) OWNER QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
4.6	Control of Configuration			
4.6.1	Contract Change Control	CRP-CN-02 CRP-EN-04 CRP-EN-05 CRP-OC-01 CRP-PR-03	Processing of Field Change Requests Processing Engineering Changes Configuration Control Board Actions Control of Modifications to Principal Project Agreements Contract and Subcontract Review Actions	
4.6.2	As-Built Verification	CRP-CN-04 CRP-QA-02 CRP-QA-13 CRP-QA-16	Construction Testing and Turnover Activity Planning Performance of Project Surveillance Inspection, Examination and Test	
4.7	Measuring and Test Equipment Calibration and Control	CRP-QA-17	Measuring and Test Equipment Calibration and Control	
4.8	Source Surveillance and Inspection	CRP-QA-02 CRP-QA-13 CRP-QA-14 CRP-QA-15 CRP-QA-16	Activity Planning Performance of Project Surveillance Processing of Responses to Nuclear Regulatory Commission Inspection Reports and Their Follow-Up During Design and Construction Arranging for Nuclear Regulatory Commission Inspections Inspection, Examination and Test	

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Figure 17A-12 (Cont'd.). Owner Quality Assurance Program Index

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
4.9	Receiving Inspection			
4.9.1	Planning and Inspection	CRP-QA-02	Activity Planning	
		CRP-QA-16	Inspection, Examination and Test	
4.9.2	Documentation	CRP-AA-04	Incoming Mail	
		CRP-AA-14	Controlled Documents	
		CRP-CN-01	Processing of Construction Data Submitted by the Constructor	
		CRP-CN-02	Processing of Field Change Requests	
		CRP-CN-04	Construction Testing and Turnover	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-OP-02	Operations Division Review and Concurrence with Engineering Design Data	
		CRP-OP-03	Operations Division Review and Concurrence with Licensing Data	
		CRP-PR-03	Contract and Subcontract Review Actions	
		CRP-QA-10	Quality Assurance Review and Approval of Engineering Documents	
		CRP-QA-11	Quality Assurance Review of Procurement Documents	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	
		CRP-CN-11	Review, Approval and Issuance of Construction Documents Non Requiring Project Office Approval	
4.9.3	Disposition of Received Items			Delegated
4.10	Control of Nonconforming Items	CRP-QA-03	Control of Nonconformances	Delegated
4.11	Control of Received Items			See Section 8
4.12	Quality Audits			
5.	Manufacturing, Fabrication and Assembly			Delegated

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FIGURE 17A-12 (Cont'd.). OWNER QUALITY ASSURANCE PROGRAM INDEX

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REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
I	Organization		Quality Assurance Program Description	
			Quality Assurance Charter	
		CRP-DR-02	Organization Plan and Functional Statements	
		CRP-QA-24	Personnel Indoctrination	
		CRP-QA-25	Administration of Personnel Certification and Records	
		CRP-QA-26	Personnel Certification	
II	Program		Quality Assurance Program Description	
			Quality Assurance Charter	
		CRP-AA-01	Management Procedures	
		CRP-AA-06	Centralized Action Correspondence Control System	
		CRP-AA-07	Reports Control Program	
		CRP-AA-11	Control of Project Office Procedures Manual	
		CRP-AA-14	Controlled Documents	
		CRP-OC-01	Control of Modifications to Principal Project Agreements	
		CRP-EN-09	Preparation and Maintenance of the Project Level 1 Schedule	
		CRP-PC-02	Preparation and Maintenance of the CRBRP Project Level 0 Schedule	
		CRP-PC-03	CRBRP Project Monthly Progress Report	
		CRP-PC-05	Preparation and Maintenance of the Work Breakdown Structure (WBS)	
		CRP-PC-06	Revision of the CRBRP Project Management Policies and Requirements	

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Figure 17A-13 OWNER QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
III	Design Control	CRP-QA-01	Quality Assurance Program Management Review Meetings	
		CRP-QA-08	Quality Assurance Program Progress and Status Review and Reporting	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	
		CRP-QA-14	Processing of Responses to Nuclear Regulatory Commission Inspection Reports and Their Follow-up During Design and Construction	
		CRP-QA-15	Arranging for Nuclear Regulatory Commission Inspections	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
		CRP-CN-01	Processing of Construction Data Submitted by the Constructor	
		CRP-EN-01	Design Control	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-EN-03	Design Reviews	
		CRP-EN-04	Processing Engineering Changes	
		CRP-EN-05	Configuration Control Board Actions	
		CRP-EN-07	Technical Control of CRBRP Test Programs	
		CRP-EN-11	Review, Approval, and Issuance of Construction Documentation Requiring Project Office Approval	

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FIGURE 17A-13 (Cont'd.) OWNER QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
		CRP-EN-09	Preparation and Maintenance of the Project Level 1 Schedule	
		CRP-EN-10	Design Document Control	
		CRP-OP-02	Operations Division Review and Concurrence with Engineering Design Data	
		CRP-PS-01	Preparation of Amendments to and Maintenance of the PSAR and ER	
		CRP-PS-02	Preparation of Responses to Nuclear Regulatory Commission Questions	
		CRP-PS-03	Preparation and Approval of Responses to Requests for Licensing and Safety Information	
		CRP-QA-10	Quality Assurance Review and Approval of Engineering Documents	
IV	Procurement Document Control	CRP-CN-01	Processing of Construction Data Submitted by the Constructor	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-PR-03	Contract and Subcontract Review Actions	
		CRP-QA-11	Quality Assurance Review of Procurement Documents	

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Figure 17A-13 (Cont'd) OWNER QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
V	Instructions, Procedures and Drawings	CRP-AA-01	Management Procedures	
		CRP-AA-03	Preparation Correspondence	
		CRP-AA-11	Control of the Project Office Procedures Manual	
		CRP-AA-14	Controlled Documents	
		CRP-PC-06	Revision to the CRBRP Project Management Policies and Requirements	
		CRP-PS-01	Preparation of Amendments to and Maintenance of the PSAR and ER	
		CRP-EN-11	Review, Approval, and Issuance of Construction Documentation Requiring Project Office Approval	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
VI	Document Control	CRP-AA-01	Management Procedures	
		CRP-AA-02	Filing Procedure for Official Project Files	
		CRP-AA-03	Preparation of Correspondence	
		CRP-AA-04	Incoming Mail	
		CRP-AA-11	Control of the Project Office Procedures Manual	
		CRP-AA-07	Reports Control Program	
		CRP-AA-14	Controlled Documents	
		CRP-PC-06	Revision to the CRBRP Project Management Policies and Requirements	
		CRP-PS-01	Preparation of Amendments to and Maintenance of the PSAR and ER	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
		CRP-EN-11	Review, Approval, and Issuance of Construction Documentation Requiring Project Office Approval	
		CRP-CN-01	Processing of Construction Data Submitted by the Constructor	
		CRP-EN-01	Design Control	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-EN-10	Design Document Control	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CFR 50, APPENDIX B

FIGURE 17A-13 (Cont'd) OWNER QUALITY ASSURANCE PROGRAM INDEX

ATTACHMENT 1

QUALITY ASSURANCE
PROCEDURE DESCRIPTIONS

Management Procedures (CRP-AA-01)

This procedure defines the responsibilities and actions for the preparation, review, approval, distribution and revision of CRBRP Project Office procedures. This procedure establishes the framework for the dissemination of basic policies, information and procedural practices.

Filing Procedure for Official Project Files (CRP-AA-02)

This procedure defines the actions and responsibilities for establishing a Project Office-wide file identification, collection, maintenance and retrieval system. The procedure details the particulars for handling and filing all official Project documents.

Preparation of Correspondence (CRP-AA-03)

This procedure describes the approved format for the preparation and handling of Project Office correspondence.

Incoming Mail (CRP-AA-04)

This procedure defines the actions and responsibilities for receipt and control of mail incoming to the Project Office. The procedure also describes the measures to be effected for preparation, distribution and maintenance of controlled documents.

Centralized Action Correspondence Control System (CRP-AA-06)

This procedure covers the establishment of action correspondence identification and control and the access of action correspondence status information.

Reports Control Program (CRP-AA-07)

This procedure defines the actions and responsibilities for the evaluation of the usefulness of inter-Project Office reports. The procedure also describes the process for approval of new reports and establishes a Project Office Reports Directory.

Control of Project Office Procedures Manual (CRP-AA-11)

This procedure defines the actions and responsibilities for the preparation, maintenance, and control of the Project Office Procedures Manual which contains copies of all Project Office Procedures including those for quality assurance activities.

Controlled Documents (CRP-AA-14)

This procedure defines the responsibilities and actions of Project Office Divisions for controlled documents generated by Project participants.

Processing of Construction Data Submitted by the Constructor (CRP-CN-01)

This procedure defines the actions and responsibilities associated with internal review and/or approval of data submitted by the Constructor.

Processing of Field Change Requests (CRP-CN-02)

This procedure defines the actions and responsibilities for control of field change notices for the construction base line. The procedure also describes the Project Office review process and coordination for approvals.

Construction Testing and Turnover (CRP-CN-04)

This procedure defines the actions and responsibilities of the Project Office to effect an orderly transfer of custody from the constructor to the operator for pre-operational and start-up activities.

Changes in Project Scope or Major Deviations from The Reference Design (CRP-DR-01)

This procedure defines the actions and responsibilities for the identification of major changes in scope or deviations from reference design. The procedure also describes the process whereby the Project Steering Committee passes such information to the Board of Directors of the Project Management Corporation.

Organization Plan and Functional Statements (CRP-DR-02)

This procedure defines the actions and responsibilities for maintenance of functional statements for Project Office organization elements, delegations of authority and organization charts. The procedure also describes the process for securing approval for proposed changes.

Design Control (CRP-EN-01)

This procedure defines the actions and responsibilites necessary to provide for a planned, disciplined approach to CRBRP Project Office control of design data.

Processing Principal Design Documents (CRP-EN-02)

This procedure defines the actions and responsibilities for conduct of internal reviews, reconciliation of comments and final action on Data Type 1 documents. This procedure also covers the internal review of Data Type 2, 3 and 4 documents submitted for information.

Design Reviews (CRP-EN-03)

This procedure defines the actions and responsibilities associated with the Project Office monitoring of formal design reviews conducted by the design contractor. This procedure also describes the documentation of design review activity by the Project Office.

Processing Engineering Changes (CRP-EN-04)

This procedure defines the actions and responsibilities for review and approval of Class 1 Engineering Change Proposals (ECP) through the mechanics of a Configuration Control Board (CCB).

Configuration Control Board Actions (CRP-EN-05)

This procedure defines the actions and responsibilities of the Configuration Control Board in handling and disposing of Engineering Change Proposals (ECP).

Development Program Technical Management (CRP-EN-06)

This procedure defines the actions and responsibilities for managing and coordinating the CRBRP contributing and applied base development programs. The procedure also describes the process of Project Office approval of Development Activity Descriptions (DADs) and Development Requirement Specifications (DRS).

Technical Control of CRBRP Test Programs (CRP-EN-07)

This procedure defines the actions and responsibilities for planning and performing preoperational, start-up, surveillance and inservice inspection testing of CRBRP systems and components. The procedure also describes the necessary documentation requirements.

Preparation and Maintenance of the Project Level 1 Schedule (CRP-EN-09)

This procedure defines the actions and responsibilities for preparing and maintaining the Level 1 schedule. The procedure also describes the processes for monitoring, controlling, and documenting progress and changes.

Design Document Control (CRP-EN-10)

This procedure defines the actions and responsibilities for control of principal design documents to assure use of current, approved design information. The procedure also describes the handling system for other reference design and guidance documents.

Review, Approval and Issuance of Construction Documentation Requiring Project Office Approval (CRP-EN-11)

This procedure defines the actions and responsibilities for the review, approval and issuance of baselined construction documentation.

Control of Modifications to Principal Project Agreements (CRP-OC-01)

This procedure defines the actions and responsibilities for the administration of the four party agreements. The procedure specifically addresses the process of change control.

Operations Division Review and Concurrence with Engineering Design Data (CRP-OP-02)

This procedure defines the actions and responsibilities for the review and concurrence for engineering data by the Project Office Operations Division. The procedure also describes the method used by the Operations Division for securing TVA Operations review and comments.

Preparation and Maintenance of the CRBRP Project Level 0 Schedule (CRP-PC-02)

This procedure defines the actions and responsibilities for preparation, review, approval and maintenance of the Project Level 0 Schedule.

CRBRP Project Monthly Progress Report (CRP-PC-03)

This procedure defines the actions and responsibilities for the preparation, review, approval and distribution of the Project Monthly Progress Report.

Preparation and Maintenance of the Work Breakdown Structures (WBS) (CRP-PC-05)

This procedure defines the actions and responsibilities for preparation and maintenance of the Project Work Breakdown Structure (PWBS); the review and integration of contractor extension of structures (CWBS) as well as the WBS Dictionary.

Revision to the CRBRP Project Management Policies and Requirements (CRP-PC-06)

This procedure defines the actions and responsibilities for the preparation and maintenance of all sections of the Management Policies and Requirements Manual. The procedure also describes the actions for coordination, review and approval of amendments thereto.

Contract and Subcontract Review Actions (CRP-PR-03)

This procedure establishes the framework for Project Office implementation of required DOE procurement regulations.

Preparation of Amendments to and Maintenance of the PSAR and ER (CRP-PS-01)

This procedure defines the actions and responsibilities for preparation of periodic amendments to both the PSAR and ER. The procedure also details the actions for coordination, review and approval of PSAR or ER documentation forwarded to NRC.

Preparation of Responses to Nuclear Regulatory Commission Questions (CRP-PS-02)

This procedure defines the actions and responsibiities for review, coordination and preparation of Project Office responses to NRC questions on the PSAR. This procedure provides detailed guidance for the technical response and priority of action.

Preparation and Approval of Responses to Request for Licensing and Safety Information (CRP-PS-03)

This procedure defines the actions and responsibilities for evaluating requests for licensing and/or safety information. The procedure also details the steps for preparation, review and approval of responses prepared for NRC or other agencies formally requesting information.

Maintenance of Consistency Between the PSAR and the Project Baseline Documentation (CRP-PS-05)

This procedure defines the actions and responsibilities for the correction of inconsistencies in the PSAR with all Project baseline documentation and other errors in the PSAR.

Distribution, Evaluation, and Preparation of Responses to NRC's Office of Inspection, and Enforcement Bulletins, Circulars, and Notices (CRP-PS-06)

This procedure defines the actions and responsibilities for the review, coordination and preparation of Project Office responses to bulletins, circulars, or notices issued by the NRC.

Quality Assurance Program Management Review Meetings (CRP-QA-01)

This procedure defines the responsibilities and actions for planning, scheduling, conducting and the reporting of results of the quarterly Project-wide quality assurance program management review meetings.

Activity Planning (CRP-QA-02)

This procedure defines the responsibilities and actions for the preparation of plans for the conduct of discrete quality assurance activities. The review of these activities in accordance with an activity plan is a part of the overall method for achieving quality.

Control of Nonconformances (CRP-QA-03)

This procedure defines the responsibilities and actions for the handling of nonconformances reported for items and services provided on the CRBRP Project. The procedure provides for the clearing or disposition of nonconformances observed.

Corrective Action Requests (CRP-QA-04)

This procedure defines the responsibilities and mechanics for initiating actions to correct improper work or conditions observed during the normal course of quality assurance activities.

Processing of Unusual Occurrence Reports (CRP-QA-05)

This procedure defines the responsibilities for evaluation of all Project related unusual occurrence reports. The procedure also details the actions of the Project Office Quality Assurance Division in terms of disposition.

Nonconformance, Unusual Occurrence and Corrective Action Analysis (CRP-QA-06)

This procedure defines the responsibilities for analysis of selected nonconformances, unusual occurrences, or corrective action reports for adequacy of corrective action proposed or taken. This procedure also details the processes for internal handling of these reports.

Quality Records (CRP-QA-07)

This procedure defines the actions and responsibilities for establishing, implementing, operating and maintaining a Project Office Quality Records Center. The procedure also details the criteria for selection of data documents for quality records retention.

Quality Assurance Program Progress and Status Review and Reporting (CRP-QA-08)

This procedure defines the responsibilities and actions for the review of the overall Quality Assurance Program on a monthly and semi-annual basis. These reviews result in the preparation of comprehensive program status reports for the Project Director's information and action as appropriate.

Quality Trend Analysis (CRP-QA-09)

This procedure defines the responsibilities for continuous monitoring of product and programmatic data to discern and identify quality trends. The procedure also outlines the process for initiating corrective action.

Quality Assurance Review and Approval of Engineering Documents (CRP-QA-10)

This procedure defines the responsibilities for review of engineering specifications, descriptions, drawings and change proposals for adequacy in terms of quality assurance requirements. The procedure also details the process of documenting the review findings.

Quality Assurance Review of Procurement Documents (CRP-QA-11)

This procedure defines the responsibilities for review of procurement documents for adequacy as related to quality assurance requirements. The procedure also details the process of documenting the review findings.

Review of Contractor Quality Assurance Plans and Procedures (CRP-QA-12)

This procedure defines the responsibilities for review of CRBRP participant quality assurance plans and procedures to determine their adequacy. The acceptance/approval process is also described.

Performance of Project Surveillance (CRP-QA-13)

This procedure defines the responsibilities for the planning, conduct and reporting of surveillance activities performed by the CRBRP Quality Assurance Division.

Processing of Responses to Nuclear Regulatory Commission Inspection Reports and Their Follow-up During Design and Construction (CRP-QA-14)

This procedure defines the responsibilities and actions for the review of NRC Inspection reports by the CRBRP Project Office. The procedure also details the actions for preparation and coordination of the formal CRBRP Project Office response to NRC.

Arranging for Nuclear Regulatory Commission Inspections (CRP-QA-15)

This procedure defines the responsibilities for internal handling of notification of NRC inspections that are to be conducted. The procedure also describes the communication links and coordination channels for the necessary arrangements.

Inspection, Examination and Test (CRP-QA-16)

This procedure defines the responsibilities for the preparation for and performance of quality assurance inspections, examinations and tests during design development, procurement, construction, installation, start-up, operation, maintenance and modification of the CRBRP.

Measuring and Test Equipment Calibration and Control (CRP-QA-17)

This procedure defines the actions and responsibilities for the verification that measuring and test equipment used for inspections, examinations or tests are properly calibrated and controlled.

Administration of Quality Assurance Auditing (CRP-QA-19)

This procedure defines the responsibilities for the planning, conduct, follow-up, and close out of quality assurance audits. This procedure also details the actions of the quality assurance audit administrator in documenting the audit activity.

Preparation, Maintenance and Control of Project Office Quality Assurance Manual (CRP-QA-20)

This procedure defines the actions and responsibilities for the preparation, distribution, maintenance and control of the CRBRP Quality Assurance Manual.

Conduct of Product Audits (CRP-QA-12)

This procedure defines the actions and responsibilities for the preparation, conduct and reporting of quality assurance product audits by the CRBRP Project Office. The procedure also details the actions of the audit team in the course of the evaluation of selected products for conformance to quality requirements.

Conduct of Programmatic Audits (CRP-QA-22)

This procedure defines the responsibilities for the preparation, conduct, and reporting of quality assurance programmatic audits by the CRBRP Project Office. The procedure details the actions of the audit team in the course of the evaluation of programmatic practices for conformance to the quality assurance program requirements.

Preparation, Transfer, and Receipt of Project Office Quality Records (CRP-QA-23)

This procedure defines the responsibilites and actions to be executed by each Project Office Division in the preparation and transfer of quality records to the Quality Assurance Division. The procedure also defines the responsibilities and action of the Quality Assurance Division when receiving quality records from other Project Office Divisions.

Personnel Indoctrination (CRP-QA-24)

This procedure defines the responsibilities and actions to provide for the indoctrination of CRBRP Project Office personnel who carry out duties affecting the quality of the CRBRP Plant structures, systems and components.

Administration of Personnel Certification and Records (CRP-QA-25)

This procedure defines the responsibilities for the administration of certification for Quality Assurance Division personnel directly involved in quality verification, testing, evaluation or audit activities. The procedure also details the actions associated with collection and maintenance of records pertaining to personnel certification.

Personnel Certification (CRP-QA-26)

This procedure defines the responsibilities and actions necessary to identify areas of quality importance for which qualifications or certification of personnel are required. The procedure also details the actions for verifying the adequacy of personnel training programs, certification practices and documentation.

Unusual Occurrence Report Preparation and Disposition (CRP-QA-27)

This procedure defines the actions and responsibilities for documenting an unusual occurrence observed during the course of work on the CRBRP Project. The procedure also details the action related to evaluation of the reportability of the event to NRC as well as the channels for reporting to NRC.

ATTACHMENT II

PROCEDURE RELEASE SCHEDULE

CRP Procedure Number	Title	Scheduled Release Date
CRP-CN-02	Processing of Field Change Requests	October 1, 1982

THE CLINCH RIVER BREEDER REACTOR PLANT

CHAPTER 17.0 - QUALITY ASSURANCE

APPENDIX C

A DESCRIPTION OF THE BALANCE OF PLANT SUPPLY

QUALITY ASSURANCE PROGRAM

CLINCH RIVER BREEDER REACTOR PLANT
A DESCRIPTION OF THE BALANCE OF PLANT
SUPPLY QUALITY ASSURANCE PROGRAM

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CLINCH RIVER BREEDER REACTOR PLANT
A DESCRIPTION OF THE BALANCE OF PLANT SUPPLY
QUALITY ASSURANCE PROGRAM

0.0 INTRODUCTION

0.1 SCOPE

Contained herein is a description of the plans and actions by the Balance of Plant (BOP) Supplier to assure the quality of certain Balance of Plant (BOP) components of the Clinch River Breeder Reactor Plant (CRBRP). These plans and actions constitute the BOP Supply Quality Assurance Program.

0.2 BASIS

The BOP, as defined by the Project, includes all other structures, systems and components of the plant not included in the Nuclear Steam Supply System (NSSS). The CRBRP Owner has assigned to other selected major Project participants the action of supplying a large portion of the BOP equipment. The CRBRP Owner has chosen to supply certain BOP equipment items and has elected to retain the execution responsibility for this major portion of the overall Project Quality Assurance Program. Assisting the Owner in this endeavor by performing supplier surveillance functions is Burns and Roe, Inc. This combined organization is then designated the BOP Supplier with overall responsibility for execution of the BOP supply quality assurance program.

The BOP Supply program requires that all technical documents for procurement of equipment, such as drawings, specifications and statements of requirements are provided by other participants and are not covered by program activity.

0.3 APPLICATION

The BOP Supply Quality Assurance Program described herein is applicable to the procurement, and through contract to the manufacturing, of certain CRBRP BOP equipment. It includes application to all BOP supplier furnished structures, systems, components and consumables (including those in Section 3.2, 7.1, and 9.13) whose satisfactory performance is required for the plant to operate reliably, safely, and with minimum environmental effects.

1.0 ORGANIZATION

The Owner is RESPONSIBLE for the overall management of the Project to design, build, and operate the Clinch River Breeder Reactor Plant (CRBRP). The execution of this responsibility rests with the CRBRP Project Director, who is the principal operations officer of the Owner. Part of this responsibility is to assure that the BOP equipment is designed, manufactured and supplied in a way that will provide adequate confidence that it will perform satisfactorily in service. To provide assurance that the BOP equipment will be supplied so as to perform satisfactorily in service, a quality assurance program has been established which shall have the objectives, carry out the functions, and be executed as hereafter defined.

1.1 FUNCTION

The functions that the Owner will perform in order to achieve the stated objectives of the BOP Supply Quality Assurance Program and fulfill its ultimate responsibility for program adequacy are as follows:

1. The development of an overall plan for the conduct of the BOP supply portion of the quality assurance program.

2. The development of working plans and procedures to conduct the BOP supply program activities.

3. Organizing and staffing appropriately to implement the BOP supply program activities.

4. The securing and directing of support services in the conduct of BOP supply program activities.

5. The surveillance over and management coordination of supplier programs.

6. The development and execution of program activities to meet necessary requirements.

1.2 RESPONSIBILITY AND AUTHORITY

The Manager of Quality Assurance for BOP supply is also the Chief, Quality Assurance, serving as head of the CRBRP Project Office Quality Assurance Division and reporting directly to the Projct Director. The Chief, Quality Assurance, acting as the Manager of Quality Assurance for BOP Supply, is assigned responsibility for devising, recommending establishment of and assuring effective execution of the BOP Supply Quality Assurance Program and assuring that organization, systems, and procedures at all levels will provide assurance that the BOP equipment is designed in accordance with requirements, is manufactured as designed, and is supplied in accordance with plans and procedures. In carrying out these responsibilities, he is authorized by the Project Director to:

1. Identify quality problems.

2. Initiate, recommend, or provide solutions through designated channels.

3. Verify implementation of solutions.

4. Determine the adequacy of facilities and equipment provided to carry out the approved procedures and instructions.

5. Authorize issuance of special instructions necessary to execute his responsibilities.

6. Notify responsible management of unsatisfactory work or unapproved practices and if necessary, stop unsatisfactory work or control further processing, delivery, or installation of nonconforming materials.

The Chief, Quality Assurance, is responsible for organizing the BOP Supply Quality Assurance Program and for recommending assignments of execution responsibility as appropriate. He shall secure charter statements from other participants describing responsibilities and functions. He shall assure that in each participant's organization the person responsible for quality assurance is granted sufficient authority to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions.

The Chief, Quality Assurance, is responsible for recommending to the Director the organization and staffing plan for the Quality Assurance Division and for managing the Quality Assurance Division in the Conduct of those quality assurance practices necessary to fulfill the BOP Supplier responsibilities for establishment and adequacy of the program. In this position, he is responsible for the technical and administrative control (except for DOE personnel) of individuals and groups of the Quality Assurance Division, performing quality assurance activities, or verifying adequacy in the performance of quality assurance related activities of others. The Project Director retains administrative control of DOE personnel.

The AE Project Quality Assurance Manager receives technical direction and functional assignment for the BOP Supply Program from the Chief, Quality Assurance. Administratively he and his quality assurance groups are a part of the AE internal organization where he reports directly to the Vice President, Breeder Reactor Division for quality assurance program effectiveness.

The supplier surveillance function is delegated to the AE Project Quality Assurance Section. The manufacturing functions of the program are or will be delegated to the various equipment suppliers by contract. All organizations who are delegated quality assurance functions, report their progress and quality achievement to the Chief, Quality Assurance.

1.3 ORGANIZATIONAL ARRANGEMENTS

The BOP Supplier organization for performing quality-related activities associated with management, planning and procurement of BOP equipment and the responsibility and authority of key positions within that organization are described in Section 1.4 of the PSAR for the Owner and the AE respectively. The BOP Supplier organization is shown in Figures 1.4-1 and 1.4-7 for the Owner and the AE respectively. The organization that will manage and implement the BOP Supply Quality Assurance Program is shown in Figure 17C-1 and is a combination of the Owner Quality Assurance Division and the Architect-Engineer Quality Assurance Section. The organization functions as a joint unit with support being drawn from the Owner's and the AE's quality assurance staff. The organization is subdivided along functional lines to perform quality verification, quality engineering, quality improvement, and AE quality assurance support. A description of the organizational elements are contained in subsequent paragraphs.

1.3.1 Quality Verification Branch

The function of the Quality Verification Branch is to maintain surveillance over quality assurance programs of major contractors and suppliers and to verify quality achievement in their work performance. This branch is also responsible for monitoring the BOP Supply quality assurance program to verify overall adequacy.

The Quality Verification Branch performs three types of activities as described below:

1.3.1.1 Surveillance

Monitoring of the Project work and the quality assurance practices on that work is performed through this activity. It also serves as the focal point for interface coordination between the BOP Supply quality assurance program and the quality assurance programs of other major contractors and suppliers.

The initiation, coordination and follow-up of inspections, reviews and audits is performed through this activity including making arrangements for support needed from other sections in the quality assurance organization, other staff organizations or through service contracts.

1.3.1.2 Inspection

Inspection of items and services or the monitoring of inspections of others is accomplished through this activity. This activity includes performance of selected civil, structural, electrical, mechanical and welding inspections, and nondestructive examinations.

1.3.1.3 Audit

Planning and conducting internal audits of the BOP Supplier's Quality Assurance Program and external audits of contractor Quality Assurance Programs is accomplished through this activity. Scheduled and unscheduled audits are conducted.

1.3.2 Quality Engineering Branch

The function of the Quality Engineering Branch is to plan, define, and develop the BOP Supply Quality Assurance Program. This function includes the preparation and maintenance of BOP Supply Quality Assurance Program requirements and internal plans and procedures. This branch has lead responsibility for quality assurance program progress and status reporting and quality records management.

The Quality Engineering Branch performs three types of activities as described below:

1.3.2.1 Planning

Planning, program development and documentation of plans and procedures for conduct of the BOP Supplier Quality Assurance Program are performed through this activity. A knowledge of industry and government standards and their appropriate application to the Project Quality Assurance Program is maintained.

1.3.2.2 Reports

Establishment of quality assurance program progress and status reporting requirements and their maintenance is accomplished through this activity. Collection of reports from branches of the Quality Assurance Division and the preparation of the BOP Supplier Quality Assurance Program progress and status report is also performed.

1.3.2.3 Records

Collection, filing, and maintaining of quality records is performed through this activity. The receiving, routing, and filing of working documentation within the Quality Assurance Division is performed. The quality records file will ultimately include records of the BOP Supply activities.

1.3.3 Quality Improvement Branch

The function of the Quality Improvement Branch is to provide needed training and indoctrination for the Quality Assurance Division and to coordinate Project Office and Project-wide training and indoctrination activities for personnel performing quality-related functions. This branch is also responsible for conducting activities wherein nonconformances are dispositioned and corrections to program deficiencies are made to improve quality achievements and to prevent .currence of nonconforming conditions.

The Quality Improvement Branch performs three types of activities as described below:

1.3.3.1 Nonconformance Control

Collecting unusual or abnormal occurrence reports, deviation requests, nonconformance reports, and deficiency citations, and processing them to satisfactory resolution is accomplished through this activity. A log of quality problems identified internally and by Equipment Suppliers will be maintained and corrective actions recorded.

1.3.3.2 Trend Analysis

Activities, reports (audit, inspection, progress, status) and records are monitored through this activity to identify quality problems. Problems identified are studied and actions recommended to correct the problem, to improve quality achievements, and to improve the efficiency and effectiveness of quality assurance activities.

1.3.3.3 Training and Indoctrination

Actions to acquaint Project personnel with the various elements of the quality assurance program and the practices needed to assure quality achievement are performed through this activity. The certification of quality assurance personnel qualifications is performed and personnel training and indoctrination within the Project Office and within other Project participant organizations is monitored to assure that:

1. Personnel performing activities affecting quality are appropriately trained in the principles, techniques and requirements of the activity being performed.

2. Personnel performing activities affecting quality are instructed as to purpose, scope, and implementation of governing manuals, policies, and procedures.

3. Appropriate training procedures are established.

4. Indoctrination and training activities are conducted in an effective manner and achieve desired results.

5. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

6. Proficiency evaluations or tests as appropriate are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

7. Certificate of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria used to qualify personnel in each function.

8. Proficiency of personnel performing and verifying activities affecting quality is maintained through work experience or retraining with continued proficiency verified through reevaluation, reexamining, and/or recertifying in accordance with Project requirements.

1.3.4 AE Project Quality Assurance Section

The AE Project Quality Assurance Manager is head of this section which functions in conjunction with and provides services for the other three branches.

1.3.4.1 Internal Audit and Surveillance

This group performs periodic internal audits and surveillances according to the BOP Supply Program plans and schedules and special audits when directed by higher authority. This group provides audit and surveillance services for and functionally performs with the audit group of the Quality Verification Branch.

1.3.4.2 Quality Assurance Engineering

This group reviews all applicable documents and provides quality assurance input for BOP Supply technical documents, procurement documents, change requests, etc., to assure proper incorporation of quality assurance requirements. This group performs functionally with the planning function of the Quality Engineering Branch.

| 1.3.4.3 Vendor Audit and Surveillance Group

This group schedules, plans and implements vendor pre-award and in-process fabrication surveys and audits. The services of qualified AE engineers are utilized whenever the scheduled audit or survey requires such capability. The group has authority to stop unsatisfactory or unapproved practices through contractual channels by virtue of the Chief, Quality Assurance for BOP Supply charter. This group provides surveillance services for and performs functionally with the surveillance function of the Quality Verification Branch.

1.4 QUALIFICATION REQUIREMENTS FOR QUALITY ASSURANCE MANAGEMENT POSITIONS

1.4.1 QUALIFICATION REQUIREMENTS OF THE CHIEF, QUALITY ASSURANCE

The individual assigned to retain overall authority and responsibility for the BOP Supplier Quality Assurance Program is the Chief, Quality Assurance who is the functional manager for directing and managing the Quality Assurance Program. He will have the following qualifications:

Education - He shall be a graduate of a four-year accredited engineering or science college or university.

Experience -

General - He shall have a minimum of 10 years experience in quality assurance or engineering, construction, or operation activities associated with nuclear facilities or equivalent heavy industry.

Specialty - He shall possess a broad knowledge and understanding of industry and government codes, standards, and regulations defining quality assurance requirements and practices.

He shall have a broad knowledge and understanding of quality assurance methods and their application.

He shall have experience in planning, defining and performing quality assurance practices and the application of procedures.

Managerial - He shall be experienced in organizing, directing and administering an overall program of activity or a major portion of an overall program having broad scope and application.

He shall have experience in the supervision of personnel and the planning and management of other resources normally needed to conduct an extensive quality assurance program.

1.4.2 CHIEF, QUALITY VERIFICATION, QUALITY ENGINEERING, AND QUALITY IMPROVEMENT

The individuals assigned to manage Quality Verification, Quality Engineering and Quality Improvement activities will have the following qualifications:

Education - He shall be a graduate of a four-year accredited science or engineering college or university.

Experience -

General - He shall have a minimum of five years experience in quality assurance or engineering, construction, or operation activities associated with nuclear facilities or equivalent heavy industry.

Specialty - He shall have a broad understanding and knowledge of applicable industry and government codes, standards, and regulations defining quality assurance requirements and practices. He shall have experience in planning, defining, and performing quality assurance practices and the application of procedures to the area of work in which he is responsible.

Managerial - He shall be experienced in organizing, directing, administering an overall program of activity or a major portion of an overall program having broad scope and application. He shall have experience in the supervision of personnel and be capable of directing and coordinating the activities of the contractors to achieve objectives.

1.4.3 QUALITY ASSURANCE MANAGER-AE SUPPORTED SERVICES

The individual responsible for management of the AE quality assurance support services will have the following qualifications:

Education - A BS degree in Engineering/Science or an equivalent combination of education and experience is preferred.

Experience -

General - He shall have a minimum of 10 years experience in quality assurance or engineering associated with design, construction, or operation of a nuclear plant facility power generating station or heavy industry.

Specialty - He shall possess a broad knowledge and understanding on industry and government codes, standards and regulations which define quality assurance program requirements and practices. He shall have a broad knowledge and understanding of quality assurance methods and their application. He shall have experience in planning, defining and implementing quality assurance practices and the application of procedures.

Managerial - He shall have a minimum of eight years experience in the supervision of personnel and the planning and management of other resources needed to develop and operationally maintain a comprehensive quality assurance program to satisfy contractually invoked requirements.

1.4.3.1 QUALITY ASSURANCE ENGINEERING GROUP - SUPERVISOR

The individual responsible for supervising the AE quality assurance engineering function will have the following qualifications:

Education - He shall be a graduate of a four-year accredited engineering college or university.

Experience -

General - He shall have a minimum of 7 years experience in quality assurance or engineering associated with design, construction, or operation of a nuclear reactor plant facility, power generating station or heavy industry.

Specialty - He shall possess knowledge and understanding of industry and government codes, standards and regulations which define quality assurance requirements and practices. He shall be familiar with methods of application of programmatic and special quality assurance requirements in design and procurement documents. He shall be experienced in evaluating program plans, procedures and practices, and subsequently verifying conformance.

Supervisory - He shall be experienced in the supervision of technical and administrative personnel engaged in quality assurance or engineering activities.

1.4.3.2 INTERNAL AUDIT AND SURVEILLANCE GROUP - SUPERVISOR

The individual responsible for supervising the AE quality assurance internal auditing and surveillance function will have the following qualifications:

Education - He shall be a graduate of a four year accredited college or university, or be a high school graduate and have 10 years experience in quality assurance/control in lieu of a degree.

Experience -

General - He shall have a minimum of 7 years experience in quality assurance or engineering associated with design, construction, or operation of a nuclear reactor plant facility, power generating station or heavy industry.

Specialty - He shall possess knowledge and understanding of industry and government codes, standards and regulations which define quality assurance requirements and practices. He shall be experienced commensurate with the scope, complexity or special nature of the activities to be audited. He shall possess good communicative skills.

Supervisory - He shall be experienced in the supervision of technical personnel engaged in quality assurance auditing or surveillance activities.

1.4.3.3 VENDOR, AUDIT AND SURVEILLANCE GROUP - SUPERVISOR

The individual responsible for supervising the AE vendor audit and surveillance function will have the following qualifications:

Education - He shall be a graduate of a four year accredited college or university, or be a high school graduate and have 10 years experience in quality assurance/control in lieu of a degree.

Experience -

General - He shall have a minimum of 7 years experience in quality assurance engineering, inspection, supervision or testing associated with design, construction, or operation of a nuclar reactor plant facility, power generating station or heavy industry.

Specialty - He shall possess knowledge and understanding of industry and government codes, standards and regulations which define quality assurance requirements. He shall be experienced in establishing and implementing programs, plans and practices for product inspection, process surveillance, and auditing of vendor QA programs. He shall have a broad knowledge and understanding of NDE, special processes and equipment test methods, including evaluation of vendor qualifications and capabilities.

Supervisory - He shall be experienced in the supervision of technical personnel engaged in inspection and test verification, vendor surveillance or auditing activities.

1.5 COMMUNICATIONS

The free, continuous, unimpeded flow of communications both horizontally and vertically within organizations as well as between the BOP Supplier's organization and other program participants is essential. The free exchange of information between responsible individuals is also essential to the expedient execution of quality assurance activities.

To promote the flow of communications and to assure positive attention to quality problems within the BOP Supplier's Quality Assurance Program, lines of communication between the Owner and the organization of the major contractors and suppliers are established as follows:

1. Communications by Senior Management - These communications will deal with such matters as major changes in the scope of the Quality Assurance Program. Communications will be addressed to the responsible senior management official with copies to the major program participants cognizant of the subject.

2. Communications by Quality Assurance Program Management - These communications will provide a direct formal or informal exchange of information between the Owner's Quality Assurance Chief and other quality assurance managers of organizations which have a direct

interface with the Owner. Copies of such communications will be distributed to other major program participants cognizant of the subject.

3. Communications by Individuals - These communications are encouraged to identify and evaluate quality problems and to initiate, recommend, or provide solutions. Communications may also be formal or informal, the choice of which shall depend on the significance of the subject and the judgment of the individuals involved.

Communication of quality assurance related activities within the BOP Supplier's organization is promoted through:

1. Periodic staff meetings of the Project Director.

2. Monthly quality assurance program progress and status reporting.

Communication of quality assurance related activities between the BOP Supplier's organization and the organizations of the major contractors and suppliers is promoted through:

1. Quarterly management review meetings attended by the Quality Assurance Managers of the major program participants.

2. Monthly quality assurance program progress and status reporting by major contractors and suppliers.

2.0 QUALITY ASSURANCE PROGRAM

2.1 PROGRAM REQUIREMENTS

The requirements for the BOP Supplier Quality Assurance Program are contained in RDT F 2-2, 1973, with Amendments 1, 2, and 3, Quality Assurance Program Requirements. Execution of a program which meets these requirements will comply with 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. A description of the activities of this program is contained herein and is presented to demonstrate how the BOP Supplier is meeting the applicable criteria.

2.2 PROGRAM ELEMENTS

The elements of the BOP Supplier Quality Assurance Program which will be executed by the BOP Supplier organization are those identified as "Program Management" complimented by those for design and development, procurement and manufacturing shown in Figure 17C-2. This is the BOP Supplier portion of the program, and it will be executed in accordance with plans, procedures and management practices of the BOP Supplier. Figures 17C-3 and 17C-4 contain a matrix of the plans and procedures and indicates the program elements as defined in 10 CFR 50, Appendix B and RDT F 2-2 respectively, that will be implemented in accordance with specific documents. A listing of these procedures with a brief description of each is contained in Attachment 1. These documents are contained in the manuals of procedures which have been developed by the BOP Supplier organizations to implement the quality assurance policies, goals and objectives described in this program description.

2.3 PROGRAM IMPLEMENTATION

As described in Section 1.0, of this Appendix, the BOP Supplier's principal operations officer is the Project Director of the CRBRP Project. The Director has day-to-day management overview involvement in the BOP Supplier Quality Assurance Program and the execution of the program.

Responsibility for the execution of the BOP Supplier Program rests with the BOP Supplier Quality Assurance Organization. Therefore, the quality assurance program management practices are performed primarily by this Quality Assurance Organization. Other units of the BOP Supplier participant organizations provide assistance for quality assurance purposes, in the execution of program activities.

To determine program status and to evaluate program adequacy, the BOP Supplier executes an overall program review practice in the form of a regular management review of the quality assurance program to assess the adequacy of its scope, implementation and effectiveness. This review is performed by the BOP Supplier Quality Assurance organization using resources and information at its disposal.

The Project Director receives the results of these management reviews in the form of regular monthly progress and status reports and other special reports as appropriate. These reports outline the progress and status of quality assurance activities, problems and nonconformances, quality trends and results of audits. The Project Director reviews these reports and initiates whatever management action is required to improve conditions and further implement the program.

In addition to the Project Director's review and assessment of the quality assurance program, an annual review and evaluation of the Project including the quality assurance program, is performed by a select committee appointed by the Breeder Reactor Corporation (BRC). The program is also subject to periodic review by the Project Steering Committee (PSC), either by itself or by some other organization on an ad hoc basis as they may choose.

The delegation of execution responsibility for program elements is accomplished through contracts. These contracts will specify applicable requirements of 10 CFR 50, Appendix B for contractor quality assurance programs.

The BOP Supplier has established a management procedures system in which significant plans and actions are documented including those affecting quality. This system provides a mechanism wherein the policies and objectives of the Project and the Project Quality Assurance Program are defined, documented and promulgated throughout the BOP Supplier organization. Within this system, detailed procedures for mandatory actions have been prepared, approved, and issued for use as described in Section 5 of this Appendix. Each procedure is issued by a directive signed by the Project Director or by the AE Vice-President, Breeder Reactor Division requiring implementation. These procedures define both the action to be performed and the responsible person or group for performing the action. To provide positive identification and control of required procedures for quality assurance activities, manuals containing these procedures have been assembled and issued and are closely controlled by the BOP Supplier participating organizations. These manuals each contain copies as appropriate of the Quality Assurance Program implementing documents listed in Figures 17C-3 and 17C-4, including the program description contained in this Appendix to Chapter 17 of the PSAR. A brief synopsis of those procedures is contained in Attachment 1.

The Quality Assurance Manuals are controlled using a document control log which shows the distribution of each copy by copy number including the distribution of revisions. The Quality Engineering Branch of the Project Office Quality Assurance Division is responsible for this activity as well as the revision and incorporation of changes to the manual defined and approved by the Chief, Quality Assurance. The contents of the Quality Assurance Manual, including the program description contained in Appendix C to Chapter 17 of the PSAR is reviewed annually as a minimum and is updated as required to maintain it current.

In the execution of the program, should a disagreement arise from a difference of opinion between quality assurance personnel and other Project Office personnel (engineering, procurement, construction, etc.), the principals themselves try to work it out. Should they fail to resolve the differences, the heads of the respective divisions are briefed on the problem by the principals and they attempt to resolve the differences on their level. Should they fail also, the problem is presented to the Project Director by the heads of the divisions involved, and he arbitrates the matter and renders a decision.

Personnel of the Owner's organization who are assigned responsibility for verifying that contractor performance is in accordance with requirements are selected and assigned to their area of responsibility based upon experience, education, and management's assessment of their performance capabilities. They are observed for performance evaluation on a continuing basis by appropriate management. On-going training and indoctrination programs are conducted to familiarize personnel with technical objectives of the activity being monitored, the requirements that it must meet, the practices and procedures to be executed in verifying conformance to requirements, and the documentation of results.

3.0 DESIGN CONTROL

The responsibility for execution of design control practices relative to BOP equipment manufacturing will be delegated to equipment suppliers by contract.

Each equipment supplier that is assigned responsiblity for design is required by contract to exercise design control practices in accordance with specified requirements. These practices include the following, as appropriate:

- o Design Planning

- o Design Definition and Control
  1. Design Criteria
  2. Codes, Standards and Practices
  3. Engineering Studies and Analyses
  4. Parts, Materials and Processes
  5. Design Descriptions
  6. Specifications, Drawings and Instructions

7. Identification
8. Acceptance Criteria
9. Interface Control
10. Engineering Holds
11. Calculations
12. Computer Codes

- Document Review and Control
  1. Document Reviews
  2. Document Control
  3. Engineering Drawing Lists
  4. Drawing Checks

- Design Reviews
  (Required design verification for the level of design activity accomplished is to be performed prior to release for procurement, manufacture, construction, or release to another organization for use in other design activities. In all cases, the design verification is to be completed prior to relying upon the component system or structure to perform its function.)

- Development
  (Including provisions that prototype component or feature testing is performed as early as possible prior to installation of plant equipment or prior to the point when the installation would become irreversible.)

- Failure Reporting and Corrective Action

  The required practices are to include the review of design drawings and specifications by the Quality Assurance organization to assure that the documents are prepared, reviewed, and approved in accordance with internal procedures and that the documents contain the necessary quality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

The BOP Supplier monitors major suppliers' design control practices and periodically audits their practices to specified requirements to assure proper implementation and adequacy of the practice.

4.0 PROCUREMENT DOCUMENT CONTROL

The BOP Supplier has established and implemented a practice for control of procurement documents to assure that procurement functions are accomplished in accordance with the applicable codes, standards, drawings, and specifications. This practice is carried out under written procedures which provide for coordination and implementation of procurement planning and review of procurement documents such as preprocurement plans and purchase orders, and changes thereto by designated personnel to assure that these documents are complete and correct.

Procurement documents are prepared by the AE and submitted to the Owner for review and approval. This practice will include the preparation of procurement documents to contain the following:

- o Scope of Work
- o Technical Requirements
- o Quality Assurance Program Requirements
- o Right of Access
- o Special Quality Assurance Requirements
- o Documentation Control
- o Nonconformance Control
- o Transfer of Requirements to Lower Tier Participants

This practice will also include:

- o Procurement Document Review and Approval
- o Document Control (Release, Distribution and Changes)

This practice also provides:

1. That procurement documents require suppliers to have and implement a documented quality asurance program for purchased materials, equipment, and services to an extent consistent with their importance to safety and utility;

2. That the purchaser has evaluated the supplier before the award of the purchase order or contract to assure that the supplier can meet the procurement requirements; and

3. That procurement documents for spare or replacement items will be subject to controls at least equivalent to those used for the original equipment.

An integral part of the document preparation process is the provision that the originating engineer is required to prepare a list of documents required by the procurement actions including the location of the technical specification requesting the document, together with the reason for the request. This information serves as the input to a vendor document control program maintained by the AE Computer Department. This computerized file maintains a continuous record of the status of all documentation requested from the vendor.

Whenever there are no code requirements involved, the vendor is directed to send copies of all documents to the AE for collection and eventual forwarding to the Owner. Whenever code requirements are involved, the vendor is directed to comply with the code for the accumulation of documents and the storage period required by the code. At the completion of the specified code storage period, the vendor is contractually directed to contact the Owner for instructions concerning the disposition of all records. The general and special conditions to each procurement action include contractual requirements granting the procurement agency's right of access to vendor facilities and records for source inspection or audit.

When acquisition of spare or replacement parts is a separate procurement action from that involved in procurement of original equipment, the procurement action is subjected to the same system of internal technical and quality assurance review as the original component. All quality assurance requirements and acceptance criteria are of the same level as the original procurement action.

When a plant procurement document is received by the BOP Supplier, it is routed to the Assistant Director for Procurement (AD/PR). The AD/PR coordinates the BOP Supplier review in conjunction with the cognizant engineer. The review is conducted thoroughly but as promptly as possible. To conduct reviews and expedite approval of procurement documents, a checklist is used by procurement document reviewers. These checklists are rather extensive and include such check points as whether or not the procurement documents:

1. Identify the documentation (e.g., drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and material, chemical and physical test results) to be prepared, maintained, and submitted as applicable to the purchaser for review and approval.

2. Contain or reference the design basis technical requirements including the applicable regulatory requirements, components and material identification requirements, drawings, specifications, codes and industrial standards, test and inspection requirements, and special process instructions for such activities as welding, heat treating, nondestructive testing, and cleaning.

3. Identify the applicable 10 CFR 50, Appendix B requirements which must be complied with and described in the supplier's QA program.

4. Identify those records which shall be retained, controlled, maintained, or delivered to the purchaser prior to use or installation of the hardware.

5. Contain the procuring agency's right of access to supplier's facilities and records for source inspection and audit.

6. Provide for spare or replacement parts of safety-related structure, systems, and components are subject to controls at lease equivalent to those used for the original equipment.

All changes and revisions to procurement documents are subject to the same review and approval requirements as the original document.

Procurement document reviewers are specified by the AD/PR, and the cognizant engineer. Reviewers are selected on the basis of their qualifications and their ability to provide a meaningful input to a particular document. The selected BOP Supplier reviewers include as a minimum, the AD/PR of his designated representative, the head of the BOP Supplier quality assurance organization or his designated representative and the BOP Supplier Cognizant Engineer. The selected reviewers also include contract and procurement staff and, as appropriate, counsel, patent and finance representatives. The respective cognizant engineer has the principal responsibility for determining the suitability and adequacy of technical specifications included in procurement documents.

The quality assurance reviewer has the principal responsibility for determining the adequacy of the procurement documents regarding quality assurance requirements. He is trained and qualified in quality assurance practices and concepts to make this determination. This review is to determine that quality and quality assurance requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements.

The AD/PR has the principal responsibility for determining the overall adequacy of administrative, financial and contractual aspects of procurement documents.

All BOP Supplier comments from reviewers of procurement documents are made in writing.

After receiving and resolving all comments, the AD/PR prepares formal correspondence to the appropriate participant and reflects comments, approval, or notifies the participant of the reasons for disapproval.

The BOP Supplier organization both participates in and monitors the execution of this practice. Periodically, the organization audits or arranges for independent audit of this practice to assure proper implementation and adequacy.

The execution responsibility for procurement document control practices related to BOP equipment manufacturing is delegated to equipment suppliers by contract.

5.0 INSTRUCTIONS, PROCEDURES AND DRAWINGS

The BOP Supplier has prepared his procedures and instructions in accordance with procedures that prescribe the format to be followed and the identification system to be used. These procedures cover all activities of management, document review and control, procurement, surveillance activities, audits, and records management. These procedures prescribe methods for performing quality-related activities in conformance with the applicable requirements of 10 CFR 50, Appendix B.

The BOP Supplier procedures are organized under a Management Procedures System which is administrated by a procedures coordinator from within the BOP Supplier organization. The procedures coordinator, which is a staff function reporting to the Project Director, is assigned the function of controlling the issuance of procedures to assure coordination and consistency in format, content, etc. The procedure system itself is organized along divisional lines (Engineering, Procurement, Construction, Quality Assurance, Public Safety, Operations, Project Control, Administrative Services and others) which give the responsible managers the responsibilities for:

- Assuring that policies of a continuing nature are incorporated in the Management Procedures System (MPS).

- o Incorporating applicable laws, standards such as 10 CFR 50, Appendix B, Executive Orders, decisions and directives of the Project Steering Committee (PSC) into the procedures to the extent necessary to show the requirements placed upon the BOP Supplier.

- o Determining the coverage and content of management directives necessary to carry out their assigned functions, assuring the accuracy and currency of the procedures and arranging for the cancellation of those that become obsolete.

- o Approving procedures for which they are responsible. Obtaining review, comment and document concurrences by other organizational units when appropriate.

- o Submitting to the Procedures Coordinator:

  a. Draft procedures for review of format.
  b. Final procedures for Director approval, issuance and distribution.

- o Determining, with concurrence of General Counsel, what portions of procedures, if any, shall be communicated to the contractors. Furnishing to the Procedures Coordinator the names of contractor personnel to whom such material together with any appropriate supplementary explanation or instructions should be distributed.

The Procedures Coordinator is the prime control officer for BOP Supplier Procedures and as such:

- o Assures that style, format, content, terms, titles and numbering sequence conform to the requirements of the Management Procedures Control System.

- o After final approval, oversees printing and distribution of the procedures.

- o Assures that a master file containing the originals of all approved CRBRP Project Office procedures and a log of issued procedures and their revisions is maintained.

- o Assures that the index of procedures is maintained current.

- o Assures that all revisions and changes to existing procedures, are distributed promptly, according to the distribution in the master file.

Organizational Unit Managers are responsible for writing and implementing the procedures necessery for their division. General Administration procedures cover policies and procedures which apply to all employees. The Project Director approves for issuance all CRBRP Project Office procedures. The individual division procedures are approved by the responsible Division Manager and are forwarded by the Procedures Coordinator to the CRBRP Project Director for final approval.

Each new procedure or revision of existing procedure is prepared using the Management Procedures system numbering code and format.

Each division establishes steps for the review of draft procedures within the divisions. If a procedure applies to more than one division, the other divisions affected receive the draft procedure for review. A draft is sent to the Procedures Coordinator who reviews it for format, style, and numbering sequence.

The final procedure or revision of existing procedure is approved by the appropriate division manager responsible for that particular subdivision of procedures, is concurred in by the Chief, Quality Assurance and is approved by the CRBRP Project Director, and released for implementation.

Distribution of each procedure or revision of existing procedure is listed and filed with the procedure copy in the procedure master file. The register shows which revision is current.

The BOP Supplier practice for documenting, in written form, the requirements for and results of activities affecting quality is, itself, executed in accordance with document control procedures identified under Section 6.0, <u>Document Control</u>.

The BOP Supplier Quality Assurance organization both participates in and monitors the execution of this practice. Periodically, the Quality Assurance Organization audits or arran[illegible] for independent audit of this practice to assure implementation and a[illegible]

Responsibility for prescribing in instructions, procedures and drawings activities affecting quality relative to equipment manufacturing will be delegated to suppliers by contract. These activities include the assurance that procedures are established and controlled to provide for the preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant configuration.

6.0 <u>DOCUMENT CONTROL</u>

The BOP Supplier has established and implemented a document control system that fulfills the quality assurance program requirements and applies to those types of documents prepared by the BOP Supplier and identified in Sections 3, 4, and 5 of this description.

The controlled documents originated by the BOP Supplier are processed in a controlled manner to assure the following:

- o Uniformity of format of initial and subsequent issuances.
- o Proper identification as to the originator and date of origin of a document, and a mechanism for verification of the authenticity of information.
- o Positive review and approval by persons qualified to determine the correctness of the information presented and to judge its ultimate usefulness.

- o Prompt and accurate distribution, including a mechanism for receipt control, of both the original document and subsequent revisions to prevent inadvertant use of superseded material and to place documents in work areas in a timely manner.

- o Efficient revision of documents when necessary to clarify, correct, augment or up-date the content of a document, while preserving the integrity of originally approved and released information.

- o Documents are available at the location where the activity will be performed prior to commencing the work.

- o Quality Assurance requirements are properly stated, are adequate and are included prior to implementation.

Controlled documents are standardized by procedure as to identification, format, and numbering. These documents are reviewed for adequacy by Division Chiefs and/or the CRBRP Project Director, as appropriate. The Chief of the Division originating the controlled document determines the extent of necessary reviews. The draft controlled document is routed to the appropriate reviewing personnel/organizations. The quality assurance organization reviews and concurs with these documents with regard to quality assurance related aspects. Comments of reviewing personnel are resolved prior to final approval of the document. A record of the review sequence which has been accomplished is documented and retained. Changes or revisions are reviewed and approved by the same divisions that performed the original review and approval. If the controlled document will be issued only to personnel of the originating Division, the respective Division Chief may approve the document for issue upon completion of necessary reviews. If the controlled document is to be issued to personnel outside of the originating Division, the respective Division Chief secures any necessary higher level approvals. The Chief of the Division originating a controlled document establishes an appropriate minimum periodic review schedule for the approved document. The primary purpose of these reviews is to determine if changes in Project status have resulted in the need for revisions to the controlled documents.

The originating Division establishes and maintains an appropriate listing of the distribution of the document upon issue. A receipt page is attached to the transmitted controlled document which requests the person receiving the document to sign and date the page and return it to the originating Division. A designated person initials the respective distribution listing upon receipt of the signed page to reflect accomplishment of transmittal and receipt. He also reviews the Division's controlled document distribution listing at least bimonthly to follow-up on any delinquent receipt pages. This distribution listing is a master list which is updated periodically to show current revision, number distributed, location, etc. Revisions to controlled documents are systematically processed with the same procedure as the original. Changes are also reviewed and approved by the same Divisions that performed the original review and approvals.

The BOP Supplier quality assurance organization both participates in and monitors the execution of the document control system. Periodically the quality assurance organization audits or arranges for independent audit of the document control system to assure implementation and adequacy.

The responsibility for execution of document control practices relative to the manufacturing of BOP equipment is delegated to equipment suppliers by contract.

7.0 CONTROL OF PURCHASED MATERIAL, EQUIPMENT AND SERVICES

The BOP Supplier has established a practice for control of the procurement process in which material, equipment and services are purchased. The procurement practice includes the following:

- o Procurement Document Preparation, Review and Change Control
- o Selection of Procurement Sources
  1. Qualified personnel evaluate the suppliers capability to provide services and products of acceptable quality.
  2. The Quality Assurance and Engineering Personnel participate in the evaluation of those suppliers providing critical items.
  3. The evaluation of suppliers is based on one or more of the following:
     - a. The supplier's capability to comply with the elements of 10 CFR 5^, Appendix B, which are applicable to the type of material, equipment, or service being procured.
     - b. A review of previous records and performance of suppliers who have provided similar articles of the type being procured.
     - c. A survey of the supplier's facilities and quality assurance program practices to determine his capability to supply a product which meets the design, manufacturing, and quality assurance requirements. Results of these surveys are to be documented and filed at the buyer's facility. If an LCVIP letter of confirmation or the "CASE" Register is used to establish the qualifications of the supplier, the documentation on file is to identify the letter or audit used.
- o Bid-Evaluation and Award
- o Purchaser's control of Supplier's performance
  1. Supplier's Quality Assurance Program is evaluated for acceptability by qualified Quality Assurance Personnel and approved prior to release of work activities that must be performed under the program.
  2. Surveillance of suppliers during fabrication, inspection, testing, and shipment of matrials, equipment and components is planned and performed with quality assurance organization participation in accordance with written procedures to assure conformance to the purchase order requirements. Those procedures provide for:

a. Specification of the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits and surveillances which assure that the supplier complies with all quality and quality assurance requirements. Surveillance is performed on those items where verification of conformance to procurement requirements cannot be determined upon receipt.

c. For commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser.

o Acceptance of item or service

1. Receiving activities associated with supplier-furnished material, equipment and services is performed to assure that:

   a. The material, component, or equipment is properly identified and corresponds to the identifiction on the purchase document and with the receiving documentation.

   b. Inspection of the material, component or equipment, and acceptance records is performed and judged acceptable in accordance with predetermined inspection instructions prior to installation or use.

   c. Inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment are available at plant site prior to installation or use.

   d. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

   e. Nonconforming items are segregated when practicable, controlled and clearly identified until proper disposition is made.

o Corrective Action with regard to the procurement process

o Quality Assurance Records

1. The supplier furnishes the following records, as a minimum, to the purchaser.

a. Certifications that specifically identify (e.g., by the purchase order number) the purchased material or equipment and the specific procurement requirements (codes, standards, specifications, etc.) met by the items.

b. Certifications that identify any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair."

The review and approval of supplier-furnished data by qualified personnel is performed in accordance with established procedures.

Review and appropriate approval is provided for supplier-generated documentation, such as certifications, for completeness, acceptability, and conformance to contract requirements before accepting completed items. At receipt inspection, the BOP Supplier routinely or periodically validates supplier-furnished material certifications by means of independent analysis or overchecks. The receipt inspection planning defines the necessary inspections and tests and provides for inspection density adjustment depending upon source, quality performance history, lot size and other factors.

Procurement of spare or replacement materials and equipment is subject to present quality assurance program controls, to codes and standards, and to technical requirements equal to or better than the original technical requirements, or as required to preclude repetition of defects.

The responsibility for execution of practices for control of purchased materials, equipment and services related to BOP equipment manufacturing is delegated to equipment suppliers by contract.

8.0 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

The responsibility for execution of identification and control of materials, parts and components, is delegated to BOP equipment suppliers and the CRBRP site receiving organization by contract.

Each supplier, who has an assigned responsibility for materials (including consumables), parts and components including partially fabricated subassemblies is required by contract to establish and implement identification and control practices. The description of the practices should include organizational responsibility. Each supplier's identification requirements are to be determined during the initial planning stages and his practice will assure that identification of the item is maintained, both on or attached to the item and on records traceable to the item as required throughout fabrication, erection, installation, and use of the item; the item(s) can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical mill test reports; that the method and location of identification does not affect the function of quality or the item being identified; and that the correct identification of items is accomplished and verified prior to the release for fabrication, assembly, shipping and installation. These practices will be designed to preclude the use of incorrect or defective materials and parts important to the function of safety-related structures systems, and components.

The BOP Supplier monitors supplier identification and control of materials, parts and components practices and periodically audits the suppliers' practices to assure proper implementati‹n and adequacy.

9.0 CONTROL OF SPECIAL PROCESSES

The responsibility for execution of control of special processes during manufacturing of BOP equipment, is delegated to equipment suppliers by contract.

Project suppliers, who are assigned responsibility for activities where special processes are involved, are required, by contract, to establish and implement practices to assure adequate performance and control of special processes such as welding, heat treating, nondestructive examination, bonding, coating, soldering, plating, hard surfacing and cleaning. These practices will include the following elements:

- o Description of the criteria for determining those processes that are controlled as special processes. A listing of special processes, which are generally those processes where direct inspection is impossible or disadvantageous, should be provided.
- o Description of the organizational responsibilities, including those for the quality assurance organization, for qualification of special processes, equipment, and personnel.
- o Qualification of procedures, equipment, and personnel for performance of special processes in accordance with applicable codes, standards, quality assurance procedures, specifications, or supplementary requirements. The quality assurance organization is involved in the qualification activities to assure they are satisfactorily performed.
- o Special processes are performed by qualified personnel using qualified procedures and equipment with established procedures for recording evidence of acceptable accomplishment.
- o Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

The BOP Supplier monitors supplier special processes control practices and periodically audits the suppliers' practices to assure proper implementation and adequacy.

10.0 INSPECTION

The responsibility for direct inspection of items and work practices related to BOP equipment manufacturing, is delegated to equipment suppliers by contract.

The BOP Supplier organization will perform necessary inspections and will witness selected inspections to verify that suppliers are providing items that will meet specified requirements. Each supplier who is assigned responsibility for performing procurement or manufacturing activities that affect quality, is required by contract to establish and implement inspection practices of sufficient scope to be fully effective. These inspection practices will identify and verify conformance of items and services with the documented specifications, instructions, procedures and drawings for accomplishing the required activities. The inspection practice performed by either the BOP Supplier or the Equipment Supplier as appropriate will provide that:

- o Organizational responsibilities for inspection are described. Inspection personnel are independent from the individual or group performing the activity being inspected. If the individuals performing inspections are not part of the quality assurance organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure such as cost and schedule should be reviewed and found acceptable by the quality assurance organization prior to initiation of the activity.

- o Inspection procedures, instructions, and check lists contain the following:

  a. Identification of characteristics to be inspected.

  b. Identification of the individuals or groups responsible for performing the inspection operation.

  c. Acceptance and rejection criteria.

  d. A description of the method of inspection.

  e. Specification of the necessary measuring and test equipment including accuracy requirements.

- o Inspection procedures or instructions are available with necessary drawings and specifications for use prior to performing inspection operations.

- o Inspectors (including NDT personnel) are qualified in accordance with appropriate codes, standards, and company training programs, and their qualifications and qualification records are kept current.

- o Modifications, repairs, and replacements are inspected in accordance with the original design and inspection requirements or acceptable alternatives.

- o Program procedures are developed with quality assurance organization participation to provide criteria for determining the accuracy requirements of inspection equipment and the timing of required inspection, or definition of how and when inspections are performed.

- o Provisions are established that identify mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

- o The individuals or groups who perform receiving and process verification inspections at the construction site are identified.

- Provisions are established for indirect control by monitoring, processing methods, equipment, and personnel if direct inspection is not possible.

- Inspection results are documented, evaluated and their acceptability determined by a responsible individual or group.

The BOP Supplier monitors supplier inspection practices and periodically audits the suppliers practice to assure proper implementation and adequacy.

11.0 TEST CONTROL

The BOP Supplier delegates execution responsibility for testing and test control practices to equipment suppliers by contracts. The BOP Supplier will witness tests by the equipment supplier on a selective basis and may, upon limited occasions, conduct independent tests of his own. Each equipment supplier, who is assigned responsibilities for performing procurement and manufacturing activities that affect quality, is required by contract to establish required tests, including proof tests prior to installation and preoperational tests and to establish a testing control practice. The testing control practice for testing performed by either the BOP Supplier or the Equipment Supplier as appropriate will include the following elements:

- Identification of required testing to demonstrate that the item will perform satisfactorily in service and that testing activities are identified, documented and accomplished in accordance with written controlled procedures.

- Identification of criteria for determining the accuracy requirements of test equipment and criteria for determining when a test is required or how and when testing activities are performed.

- Written test procedures that incorporate or reference the requirements and acceptance limits contained in applicable design and procurement documents.

- Written test procedures that include:

  a. Instructions for testing method and test equipment and instrumentation.

  b. Provisions for the following as appropriate:

     - Calibrated instrumentation
     - Adequate and appropriate equipment
     - Trained, qualified, and licensed or certified personnel
     - Preparation, condition, and completeness of item to be tested
     - Suitable and controlled environmental conditions
     - Mandatory inspection hold points for witness by purchaser, contractor, or authorized inspector
     - Provisions for data collection and storage
     - Acceptance and rejection criteria
     - Methods of documenting or recording test data and results
     - Provisions are assuring test prerequisites have been met

- o Test results are documented, evaluated, and acceptance status identified by a qualified, responsible individual or group.

- o Modifications, repairs, and replacements are tested in accordance with the original design and testing requirements or acceptable alternates.

The BOP Supplier monitors the suppliers testing and test control and periodically audits the suppliers test control practice to assure implementation and adequacy.

12.0 <u>CONTROL OF MEASURING AND TEST EQUIPMENT</u>

The BOP Supplier delegates execution responsibility for control of measuring and test equipment to the suppliers through contract. In those instances where the BOP Supplier elects to perform or have his agent perform inspections, examinations, or tests, the measuring and test equipment will be controlled in accordance with those requirements imposed upon the supplier.

Each supplier, who is assigned responsibility for performing inspections, examinations, or tests, is required by contract to establish and implement a system for calibration and control of measuring and test equipment.

This system will include the following elements:

- o Procedures that describe the calibration technique and frequency, maintenance, and control of all measuring and test instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment which are used in the measurement, inspection, and monitoring of safety-related components, systems, and structures. The review and documented concurrence of these procedures is described and the organization responsible for these functions are identified.

- o Measuring and test equipment is identified and the calibration test data is identified as to the equipment to which it applies.

- o Measuring and test equipment is labeled or tagged or otherwise controlled to indicate the due date of the next calibration. The method of "otherwise controlled" should be described.

- o Measuring and test instruments are calibrated at specified intervals based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement.

- o An investigation conducted and documented to determine the validity of previous inspections performed when measuring and test equipment is found to be out of calibration.

- o Calibrating standards have an uncertainty (error) requirement of one-fourth to one-tenth of the tolerance of the equipment being calibrated. A greater uncertainty may be acceptable when limited by the "state-of-the-art." Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if it can be shown to be adequate for the requirements and the basis of acceptance is documented and authorized by responsible management. The management authorized to perform this function is identified.

- Measures are taken and documented to determ... the validity of previous inspections performed and the acceptability of items inspected or tested since the last calibration when measuring and test equipment is found to be out of calibration. Inspections or tests are repeated on items determined to be suspect.

- Reference and transfer standards are traceable to nationally recognized standards; or, where national standards do not exist, provisions are established to document the basis for calibration.

The BOP Supplier monitors the supplier's calibration and control of measuring and test equipment and periodically audits the suppliers practices to assure implementation and adequacy and effectiveness.

13.0 HANDLING, STORAGE AND SHIPPING

The BOP Supplier delegates execution responsibility for handling, storage and shipping practices to the suppliers through contracts.

Each supplier, who is assigned responsibility for manufacturing, fabrication or assembly, is required by contract to establish and implement practices for handling, storage and shipping of items. These practices will include the following:

- Special handling, preservation, storage, cleaning, packaging, and shipping requirements are specified and accomplished by suitably qualified individuals in accordance with predetermined work and inspection instructions.

- Procedures are prepared in accordance with design and procurement specification requirements to establish and describe controls for the cleaning, handling, storage, packaging, shipping, and preservation of materials, components, and systems to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity.

The BOP Supplier monitors the Suppliers handling, storage and shipping practices and periodically audits suppliers practices to assure implementation and adequacy.

14.0 INSPECTION, TEST AND OPERATING STATUS

The BOP Supplier delegates execution responsibility for inspection, test and operating status measures to suppliers through contracts.

Suppliers, who are assigned responsibility for manufacturing, are required to establish and implement practices to indicate the status of inspections and tests performed upon individual items throughout fabrication, assembly and test by using such marking as stamps, tags, labels, routing cards or other suitable means. These practices will include provisions for:

- Identification of the inspection, test, and operating status of structures, systems and components being known throughout fabrication and assembly.
- The application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps being controlled.
- Altering the sequence of or bypassing the required inspections, tests, and other critical or safety-related operations being controlled through documented measures under the cognizance of the QA organization. Such actions are to be subject to the same controls as the original review and approval.
- The status of nonconforming, inoperative, or malfunctioning structures, systems, or components being documented and identified to prevent inadvertent use. The organization responsible for this function is identified.

The BOP Supplier monitors the supplier's practices for indicating inspection, test and operating status and periodically audits suppliers practices to assure implementation and adequacy.

15.0 NONCONFORMING MATERIALS, PARTS OR COMPONENTS

The BOP Supplier has established and implemented practices for control, review and disposition of nonconforming materials, parts or components. These practices are designed to assure that measures are established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. The nonconformance control practice includes the following elements:

- Establish disposition responsibility
- Documentation and reporting
- Review, evaluation and disposition

All reports of deviations that are proposed for disposition in such a way that the finished item or completed service will not conform to the approved requirements are processed for approval in accordance with procedures established to provide a level of approval equivalent to the original approval of the requirements that will not be met as a result of the proposed disposition. These reports are part of the documentation required at the plant site.

Errors or deficiencies found in design and construction reported to or discovered by the Owner which could adversely affect safety-related structures, systems or components and which represent a breakdown in the quality assurance program, deficiency in final design, deficiency in construction or deviation from performance specifications are evaluated by Engineering, Public Safety and Quality Assurance for consideration as a reportable deficiency under 10 CFR 50.55(e). If it is concluded that the error or deficiency comes under this paragraph, the deficiency together with the proposed corrective action is reported to the Nuclear Regulatory Commission according to regulations. Defects or noncompliance in the plant or basic component supplied to the plant that are reported to or discovered by the Owner, which could adversely affect safety-related functions of the plant, are evaluated by Engineering, Public Safety, and Quality Assurance for consideration as a reportable deficiency under 10 CFR 21. If it is concluded that the defect or noncompliance is reportable under Part 21, the deficiency is reported to the Nuclear Regulatory Commission according to regulations. The deficiency, whether reportable or not, is further evaluated against the procedural requirements that should have prevented the occurrence. When the procedural system is deficient, the affected organization is required to take whatever steps are necessary to achieve appropriate corrective action to the system to preclude recurrence of the deficiency. The deficiency is reported within the Project via an unusual occurrence report as described in Section 16.

Each supplier, who is assigned responsibility for procurement, manufacturing, or construction of items of the BOP, is required by contract to establish and implement a practice for the control of nonconforming materials, parts or components. These nonconformance control practices will include the following elements:

- o The identification, documentation, segregation where practicable, review, disposition, and notification to affected organizations of nonconforming materials, parts, components, or services (including computer codes) is undertaken if disposition is other than to scrap.)

- o Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconformance, and the inspection requirements; and includes signature approval of the disposition. Nonconformances are corrected or resolved prior to the initiation of the preoperational test program on the item.

- o Provisions are established identifying those individuals or groups delegated the responsibility and authority to approve the dispositioning and close out of nonconforming items.

- o Nonconforming items are segregated, where practicable, from acceptable items, controlled and identified as discrepant until properly dispositioned.

- o Acceptability of rework or repair of materials, parts, components, systems, and structures is verified by reinspecting the item as originally inspected or by a method which is at least equal to the original inspection method; inspection, rework, and repair procedures are documented.

- o Nonconformance reports dispositioned "use as is," "use as repaired," or "use as modified" are made part of the inspection records and forwarded with the hardware to the Owner.

- o Nonconformance reports are periodically analyzed to show quality trends, and the significant results are forwarded to management for review and assessment.

These practices will assure that nonconforming items are reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures. They will include measures which control further processing, delivery or installation pending proper disposition of the deficiency.

The BOP Supplier Quality Assurance organization also participates in and monitors the execution of the nonconformance control practices and periodically audits or arranges for independent audit of the control practices to assure implementation and adequacy.

The responsibility to control nonconforming material, parts and components related to BOP equipment manufacturing and resulting subcontracts, is delegated to equipment suppliers by contract.

16.0 CORRECTIVE ACTION

A corrective action system has been established and implemented wherein conditions adverse to quality such as failures, nonconformances, malfunctions, deficiencies, deviations and defective material and equipment that are required for reliable and safe operation of the plant, are reported to the purchaser through nonconformance and unusual occurrence reporting procedures. Quality Assurance activities found deficient by reviews and audits of supplier's activities, are also reported to the Purchaser or by the Purchaser's representative performing such reviews and audits. The system includes the following elements:

- o Evaluation of nonconformances and determination of the need for corrective action in accordance with established procedures.
- o Prompt corrective action initiated and documented following the determination of a nonconformance to preclude the recurrence of those adverse conditions significant to quality.
- o Follow-up reviews conducted by the quality assurance organization to verify proper implementation of corrective actions and to close out the corrective action documentation in a timely manner.
- o Adverse conditions significant to quality, the cause of the conditions, and the corrective action taken are documented and reported to appropriate levels of management for review and assessment.

The system for corrective action is designed to respond to the following conditions:

- o A repetitive or recurring deficiency in nonconformance reports for the same operation or the same origin.
- o Detection of a condition which can significantly degrade the quality of a product or service.
- o Procedural or documentation deficiencies that can adversely affect product quality or required records.

- Detection of an unsatisfactory quality trend.

- Unreported deviations, failures, malfunctions or accident, or inadequate follow-up to reported items to establish corrective/preventive action.

- Ineffective procedures or documentation that govern products or systems.

- When an audit reponse has not been received in the allotted time.

- When nonconformance review board actions determine that a corrective action is required to obtain a specific corrective/preventive action.

The responsibility for execution of corrective action practices related to BOP equipment manufacturing is delegated to equipment suppliers by contract.

17.0 QUALITY ASSURANCE RECORDS

The BOP Supplier organization has established a quality assurance records system that provides for the collection, storage, and maintenance of BOP Supplier prepared records in accordance with approved records management procedures.

The BOP Supplier has delegated execution responsibility for other records preparation and initial collection, storage and maintenance to the other suppliers by contract. This includes those official records directly related to structures, systems and components of the plant that are prepared by and used in design, procurement, manufacturing, construction and operation. In progressive stages as required by code, standard, regulation and specification, these records will be turned over to the Owner.

The BOP Supplier quality assurance records system provides for coordination of records management practices as well as the collection, storage and maintenance of those records resulting from the BOP activities that are necessary to define the overall BOP Supply Program quality and provide objective evidence of quality achievement; however, the responsibility for preparation and initial collection of record copies of documents related to BOP equipment is delegated to equipment suppliers by contract. The system includes provisions that ensure:

- Records are maintained to provide documentary evidence of the quality of items and the activities affecting quality.

- Quality assurance records include operating logs; results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports, nonconforming and corrective action reports, and other associated documents.

- Records are readily identifiable and retrievable.

- Requirements and organizational responsibilities for record transmittals, retention, and maintenance subsequent to completion of work are consistent with applicable codes, standards, and procurement documents.

- Inspection and test records contain the following:

  a. A description of the type of observation.

  b. Evidence of completing and verifying a manufacturing, inspection, or test operation.

  c. The date and results of the inspection of test.

  d. Information related to nonconformances and other conditions adverse to quality.

  e. Inspector or data reorder identification.

  f. A statement as to the acceptability of the results.

  g. Action taken to resolve any discrepancies noted.

- Record storage facilities are constructed, located, and secured to prevent destruction of the records by fire, flooding, theft and deterioration by environmental conditions such as temperature or humidity.

The BOP Supplier quality assurance organization participates in and monitors the implementation of the records system. Periodically the quality assurance organization audits or arranges for independent audit of the records system to assure implementation and adequacy.

18.0 AUDITS

The BOP Supplier has established and implemented a quality assurance audit practice which is used to provide a comprehensive independent verification and evaluation, both internally and externally, of the status and adequacy of the BOP Supply quality assurance program methods, quality-related procedures and activities. This practice includes the BOP Supply Program as well as the programs of the suppliers. This practice is also designed to assure that procedures and activities are meaningful and comply with the overall Project Quality Assurance Program requirements. The practice includes the following elements:

- Planning

- Evaluation of Quality Assurance Methods

- Activity Audits

- Product Audits

- Record Audits

o Reporting and Corrective Action

The Quality Assurance audit practice was established during the Project planning and conceptual design stage with the scope and frequency of the audits planned and scheduled based upon the importance of the activities being performed with regard to its effect upon the reliable and safe operation of the plant and was initiated early enough to assure effective quality assurance practices during design, procurement and manufacturing, inspection and test. Audits are planned in a general way on an annual basis with a more detailed plan and schedule prepared and issued on a quarterly basis. Audits are planned to cover the evaluation of internal practices of the BOP Supplier as well as the practices of major equipment suppliers who have a direct interface with the BOP Supplier. The audit plan for each major equipment supplier's program is designed to include an objective evaluation of quality-related practices, procedures and instructions; the effectiveness of implementation; and the conformance with policy directives. These audits include the evaluation of work areas; activities including personnel training and indoctrination; process and items; and the review of documents and records to ensure that the quality assurance programs are effective and properly implemented. In each major equipment supplier's program elements of interface control are evaluated with respect to each suppliers internal activities as well as interfacing activities with his customer and his suppliers. The audit plan is supplemented by unscheduled audits where the need becomes evident.

Each audit is conducted according to pre-established written procedures and check lists that include a detailed plan for the audit with a prepared check list of items to be investigated; a meeting with responsible management personnel before the audit to review scope, purpose, and schedule of the audit and at the conclusion, to review audit findings with management having responsibility in the area audited. The need for corrective actions is established and the audit results are documented in a formal report. Each audit is conducted by trained personnel that do not have direct reponsibilities in the areas being audited.

Responsible management is then required to take the necessary action to correct the deficiencies revealed by the audit and to provide the auditing organization with a statement of proposed and completed corrective action taken. Deficient areas are monitored and promptly re-audited when necessary until corrections have been accomplished.

Audit summaries are provided in both internal and external monthly progress and status reports. The audit reports and data are analyzed by the quality assurance organization for quality trends indicating any quality problems and the effectiveness of the quality assurance program, including the need for reaudit of deficient areas. These reports are provided to management for review and assessment.

The BOP Supply audit practice is performed as a minimum in those areas of the overall Project Quality Assurance program where the requirements of 10 CFR 50, Appendix B are being implemented. Each supplier as appropriate, prepares a matrix showing which procedures are used for implementation of each of the eighteen criteria of Appendix B. The activities and practices which carry out these procedures are audited on a pre-scheduled basis. These activities include:

1. The preparation, review, approval, and control of the PSAR, designs, specifications, procurement documents, instructions, procedures and drawings.

2. The determination of site features which affect plant safety.

3. Request for proposals and evaluations of bids.

4. Indoctrination and training programs.

5. Interface control among the applicant and the principal contractors.

6. Calibration and nonconformance control systems.

7. SAR and SSAR commitments.

8. Activities associated with computer codes.

The responsibility for execution of audit practices related to BOP equipment manufacturing is delegated to equipment suppliers by contract.

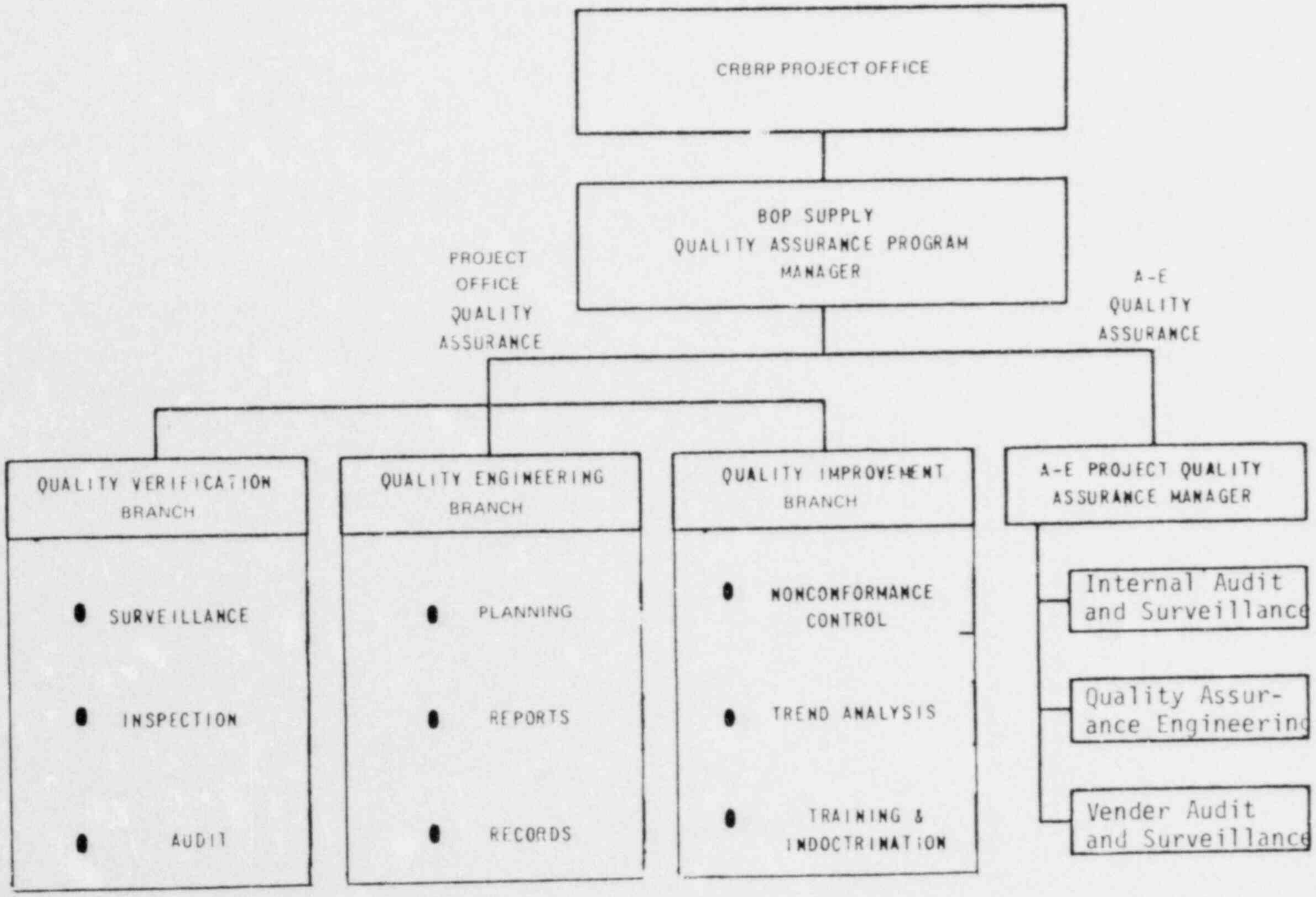

FIGURE 17C-1 BOP SUPPLIER QUALITY ASSURANCE ORGANIZATION

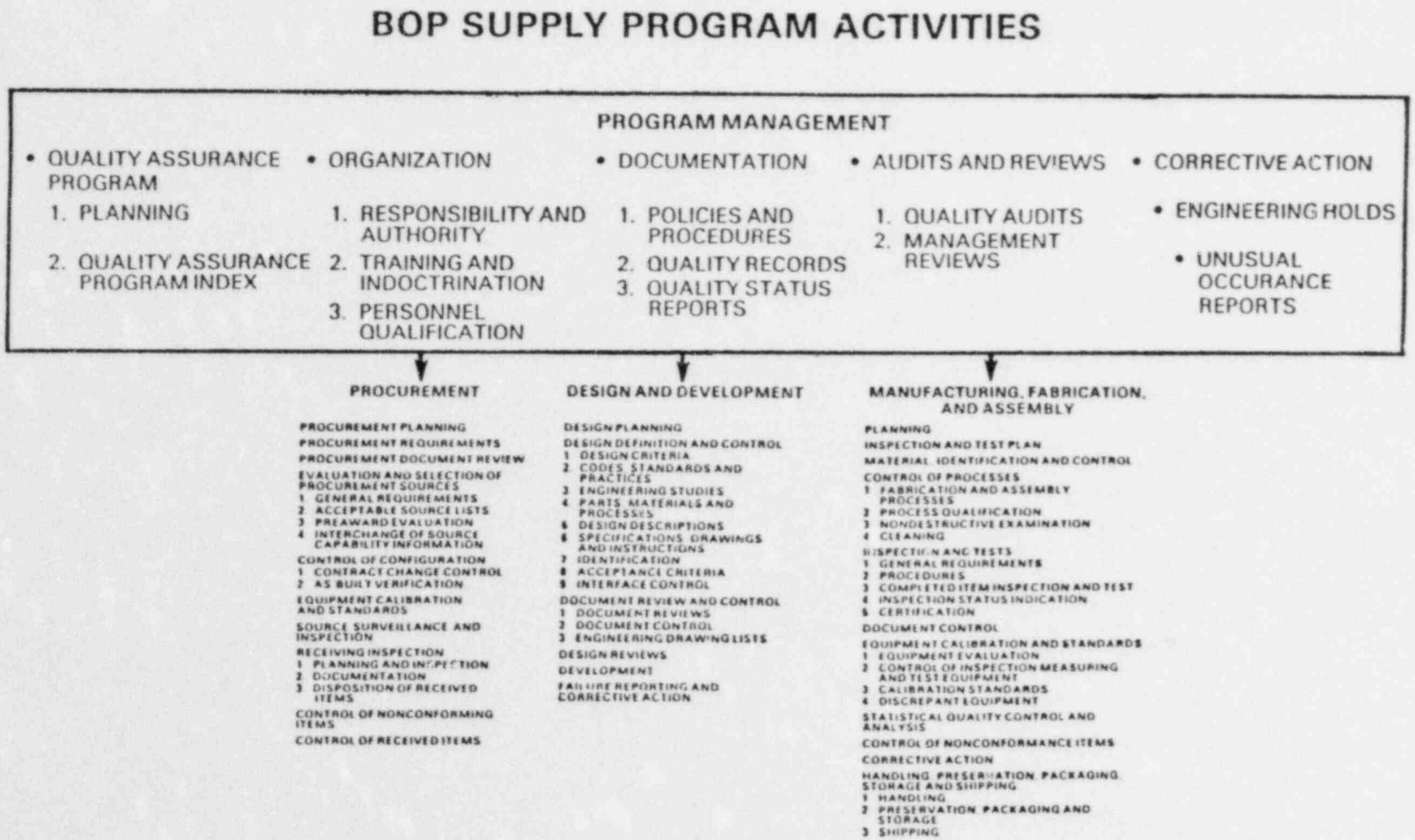

FIGURE 17C-2 MAJOR ELEMENTS OF THE BOP SUPPLY QUALITY ASSURANCE PROGRAM

REQUIREMENT OF 10CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
I	Organization		BCP Supply Program Description	
			CRBRP Quality Assurance Charter	
		CRP-DR-02	Organization Plan and Functional Statements	
		CRP-QA-24	Personnel Indoctrination	
		CRP-QA-25	Administration of Personnel Certification and Records	
		CRP-QA-26	Personnel Certification	
II	Program		BOP Supply Program Description	
			CRBRP Quality Assurance Charter	
		CRP-AA-01	Management Procedures	
		CRP-AA-06	Centralized Action Correspondence Control System	
		CRP-AA-07	Report Control Program	
		CRP-AA-11	Control of Project Office Procedures Manual	
		CRP-AA-14	Controlled Documents	
		CRP-PC-03	CRBRP Project Monthly Progress Report	
		CRP-QA-01	Quality Assurance Program Management Review Meetings	
		CRP-QA-08	Quality Assurance Program Progress and Status Review and Reporting	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
		BRD-PC-7.1	Indoctrination and Training	
III	Design Control	BRD-QA-1.18	Training and Certification of QA Personnel	Delegated
IV	Procurement Document Control			
		CRP-EN-02	Processing Principal Design Documents	
		CRP-PR-02	TVA Purchases of CRBRP Items	
		CRP-QA-11	Quality Assurance Review of Procurement Documents	
		BRD-E-2.4	Vendor/Contractor Documents	
		BRD-QA-1.21	Bid Review for Quality Requirements	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CPR 50, APPENDIX B

Figure 17C-3 BALANCE OF PLANT (BOP) SUPPLY QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
V	Instructions, Procedures and Drawings	CRP-AA-01	Management Procedures	
		CRP-AA-03	Preparation of Correspondence	
		CRP-AA-11	Control of Project Office Procedures Manual	
		CRP-AA-14	Controlled Documents	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
		BRD-PC-1.5	Procedure Preparation	
		BRD-PC-3.6	Distribution	
		BRD-QA-1.2	Preparation, Control and Distribution of QA Instructions	
		BRD-QA-1.3	Preparation of QA Procedures	
		BRD-QA-1.19	Procedure Writing Format	
		BRD-QA-3.1000	Project Surveillance	
		BRD-QA-3.1000-1	Preparation of Project Surveillance/Acceptance Checklists, Summary & Report Forms	
VI	Document Control	CRP-AA-01	Management Procedures	
		CRP-AA-02	Filing Procedure for Official Project Files	
		CRP-AA-03	Preparation of Correspondence	
		CRP-AA-04	Incoming Mail	
		CRP-AA-11	Control of Project Office Procedures Manual	
		CRP-AA-07	Reports Control Program	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-PR-02	TVA Purchases of CRBRP Items	
		CRP-QA-20	Preparation, Maintenance and Control of Project Office Quality Assurance Manual	
		BRD-E-2.4	Vendor/Contractor Documents	
		BRD-PC-1.5	Procedure Preparation	
		BRD-PC-3.6	Distribution	
VII	Control of Purchased Materials, Equipment and Service	CRP-AA-03	Preparation of Correspondence	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CFR 50, APPENDIX B

Figure 17C-3 (Cont'd) BALANCE OF PLANT (BOP) SUPPLY QUALITY ASSURANCE PROGRAM INDEX

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
		CRP-AA-04	Incoming Mail	
		CRP-LN-02	Processing Principal Design Documents	
		CRP-PR-02	TVA Purchases of CRBNP Items	
		CRP-QA-01	Quality Assurance Program Management Review Meetings	
		CRP-QA-02	Activity Planning	
		CRP-QA-07	Quality Records	
		CRP-QA-08	Quality Assurance Program Progress and Status Review and Reporting	
		CRP-QA-10	Quality Assurance Review and Approval of Engineering Documents	
		CRP-QA-11	Quality Assurance Review of Procurement Documents	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	
		CRP-QA-13	Performance of Project Surveillance	
		CRP-QA-16	Inspection, Examination and Test	
		CRP-QA-18	Quality Assurance Participation in Preparation of Approved Source Lists	
		BRD-E-2.2	Prequalification of Bidders	
		BRD-E-2.3	Technical Evaluation of Bids	
		BRD-E-2.4	Vendor/Contractor Documents	
		BRD-QA-1.11	Vendor Quality Assurance Prequalification Program	
		BRD-QA-1.11-1	Evaluation of Prequalification Questionnaire	
		BRD-QA-1.12	Vendor QA Qualification Survey	
		BRD-QA-1.12-1	Performance, Evaluation and Reporting of Preaward Surveys	
		BRD-QA-1.16	QA Review of Submittals	
		BRD-QA-1.16-1	Review of Design/Document Submittals	
		BRD-QA-1.21	Bid Review of Quality Requirements	
		BRD-QA-1.25	Nonconformance Review Board (NRB)	
		BRD-QA-3.101	Source Surveillance	
		BRD-QA-3.101-2	Administration of the Source Surveillance Program	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CFR 50, APPENDIX B

Figure 17C-3 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
		BRD-QA-3.101-3 BRD-QA-3.101-4	Preparation for and Performance of Surveillances Preparation and Issuance of Source Surveillances	
VIII	Identification and Control of Materials, Parts and Components			
IX	Control of Special Processes			Delegated
X	Inspection	CRP-QA-02 CRP-QA-16 BRD-QA-3.101 BRD-QA-3.101-4	Activity Planning Inspection, Examination and Test Source Surveillance Source Surveillance Planning	
XI	Test Control	CRP-QA-02 CRP-QA-13 CRP-QA-16 BRD-E-2.4 BRD-QA-1.16	Activity Planning Performance of Project Surveillance Inspection, Examination and Test Vendor/Contractor Documents QA Review of Submittals	
XII	Control of Measuring and Test Equipment	CRP-QA-17	Measuring and Test Equipment Calibration and Control	
XIII	Handling, Storage and Shipping			Delegated
XIV	Inspection, Test and Operating Status	BRD-QA-3.101-2	Administration of the Source Surveillance Program	
QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CFR 50, APPENDIX B				

Figure 17C-3 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
XV	Nonconforming Materials Parts and Components	CRP-AA-04	Incoming Mail	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-OP-02	Operations Division Review and Concurrence with Engineering Design Data	
		CRP-PR-02	TVA Purchases of CRBRP Items	
		CRP-QA-03	Control of Nonconformances	
		CRP-QA-05	Processing of Unusual Occurrence Reports	
		CRP-QA-27	Unusual Occurrence Report Preparation and Disposition	
		BRD-E-2.5	Vendor/Contractor Waiver Requests	
		BRD-QA-1.13	Corrective Action Request (CAR)	
		BRD-QA-1.25	Nonconformance Review Board (NRB)	
		BRD-QA-1.1000	Deviation Reporting and Control	
		BRD-L-2.3	Reporting of Defects and Noncompliance	
XVI	Corrective Action	CRP-EN-02	Processing Principal Design Documents	
			Engineering Design Data	
		CRP-QA-03	Control of Nonconformances	
		CRP-QA-04	Corrective Action Requests	
		CRP-QA-05	Processing of Unusual Occurrence Reports	
		CRP-QA-06	Nonconformance, Unusual Occurrence and Corrective Action Analysis	
		CRP-QA-09	Quality Trend Analysis	
		CRP-QA-27	Unusual Occurrence Report Preparation and Disposition	
		BRD-QA-1.13	Corrective Action Request (CAR)	
		BRD-QA-1.25	Nonconformance Review Board (NRB)	
		BRD-QA-1.1000	Deviation Reporting and Control	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CFR 50, APPENDIX B

Figure 17C-3 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF 10 CFR 50 APPENDIX B		IMPLEMENTING DOCUMENT		REMARKS
CRITERION	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
XVII	Quality Assurance Records	CRP-AA-02	Filing Procedure for Official Project Files	
		CRP-AA-04	Incoming Mail	
		CRP-QA-07	Quality Records	
		BRD-QA-1.18	Training and Certification of QA Personnel	
		BRD-PC-3.1	Filing	
		BRD-E-2.4	Vendor/Contractor Documents	
XVIII	Audits	CRP-QA-19	Administration of Quality Assurance Auditing	
		CRP-QA-21	Conduct of Product Audits	
		CRP-QA-22	Conduct of Programmatic Audits	
		BRD-QA-1.13	Corrective Action Request (CAR)	
		BRD-QA-4.2	Audits	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF 10 CFR 50, APPENDIX B

Figure 17C-3 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
4.6	Control of Configuration			Partially Delegated
4.6.1	Contract Change Control			
4.6.2	As-Built Verification	CRP-QA-02	Activity Planning	
		CRP-QA-13	Performance of Project Surveillance	
		CRP-QA-16	Inspection, Examination and Test	
		BRD-QA-3.101	Source Surveillance	
		BRD-QA-3.101-2	Administrative of the Source Surveillance Program	
4.7	Measuring and Test Equipment Calibration and Control	CRP-QA-17	Measuring and Test Equipment Calibration and Control	
4.8	Source Surveillance and Inspection	CRP-QA-02	Activity Planning	
		CRP-QA-13	Perforamnce of Project Surveillance	
		CRP-QA-16	Inspection, Examination and Test	
		BRD-QA-3.101	Source Surveillance	
		BRD-QA-3.101-4	Preparation and Issuance of Source Surveillance Reports	
		BRD-QA-3.1000	Project Surveillance	
		BRD-QA-3.1000-1	Preparation of Project Surveillance/Acceptance Check-lists, Summary and Report Forms	
		BRD-QA-3.101-3	Preparation for and Performance of Surveillances	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2

Figure 17C-4 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
4.9	Receiving Inspection			
4.9.1	Planning and Inspection	CRP-QA-02	Activity Planning	
		CRP-QA-16	Inspection, Examination and Test	
4.9.2	Documentation	CRP-AA-04	Incoming Mail	
		CRP-AA-14	Controlled Documents	
		CRP-EN-02	Processing Principal Design Documents	
		CRP-OP-02	Operations Division Review and Concurrence with Engineering Design Data	
		CRP-PR-02	TVA Purchases of CRBRP Items	
		CRP-QA-10	Quality Assurance Review and Approval of Engineering Documents	
		CRP-QA-12	Review of Contractor Quality Assurance Plans and Procedures	
		BRD-E-2.4	Vendor/Contractor Documents	
		BRD-QA-1.16	Quality Assurance Review of Submittals	
		BRD-QA-1.16-1	Review of Design/Document Submittals	
4.9.3	Disposition of Received Items			
4.10	Control of Noncorforming Items	CRP-QA-03	Control of Nonconformances	Delegated
		BRD-E-2.4	Vendor/Contractor Documents	
		BRD-QA-1.25	Nonconformance Review Board (NRB)	
		BRD-QA-1.1000	Deviation Reporting and Control	
		BRD-E-2.5	Vendor/Contract Waiver Requests	
		BRD-L-2.3	Reporting of Defects and Noncompliances	
		BRD-QA-1.13	Corrective Action Request (CAR)	
4.11	Control of Received Items			Delegated
4.12	Quality Audits			See Section 8
QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2				

Figure 17C-4 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
5.	Manufacturing, Fabrication and Assembly			Delegated
6.	Construction and Installation			Not Applicable
7.	Operation, Maintenance and Modification			Not Applicable

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2

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Figure 17C-4 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF RDT F 2-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
8.	Quality Assurance Audits			
8.2	Planning	CRP-QA-19	Administration of Quality Assurance Auditing	
8.3	Evaluation of Quality Assurance Methods	CRP-QA-22 BRD-QA-4.2	Conduct of Programmatic Audits Audits	
8.4	Activity Audits	CRP-QA-22 BRD-QA-4.2	Conduct of Programmatic Audits Audits	
8.5	Product Audits	CRP-QA-21	Conduct of Product Audits	
8.6	Nondestructive Examination Audits	CRP-QA-22 BRD-QA-4.2	Conduct of Programmatic Audits Audits	
8.7	Record Audits	CRP-QA-21 CRP-QA-22 BPD-QA-4.2	Conduct of Product Audits Conduct of Programmatic Audits Audits	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2

Figure 17C-4 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

REQUIREMENT OF RDT F-2		IMPLEMENTING DOCUMENT		REMARKS
SECTION NUMBER	TITLE	DOC. NO.	TITLE	INSTRUCTIONS REF. DOC., ETC.
8.8	Reporting and Corrective Action	CRP-QA-21	Conduct of Product Audits	
		CRP-QA-22	Conduct of Programmatic Audits	
		BRD-QA-1.13	Corrective Action Request (CAR)	
		BRD-QA-1.1000	Deviation Reporting and Control	
		BRD-QA-4.2	Audits	

QUALITY ASSURANCE PROGRAM INDEX VERSUS REQUIREMENTS OF RDT F 2-2

Figure 17C-4 (Cont'd.). Balance of Plant (BOP) Supply Quality Assurance Program Index

ATTACHMENT 1

QUALITY ASSURANCE
PROCEDURE DESCRIPTIONS

Management Procedures (CRP-AA-01)

This procedure defines the responsibilities and actions for the preparation, review, approval, distribution and revision of CRBRP Project Office procedures. This procedure establishes the framework for the dissemination of basic policies, information and procedural practices.

Filing Procedure for Official Project Files (CRP-AA-02)

This procedure defines the actions and responsibilities for establishing a Project Office-wide file identification, collection, maintenance and retrieval system. The procedure details the particulars for handling and filing all official Project documents.

Preparation of Correspondence (CRP-AA-03)

This procedure describes the approved format for the preparation and handling of Project Office correspondence.

Incoming Mail (CRP-AA-04)

This procedure defines the actions and responsibilities for receipt and control of mail incoming to the Project Office. The procedure also describes the measures to be effected for preparation, distribution and maintenance of controlled documents.

Centralized Action Correspondence Control System (CRP-AA-06)

This procedure covers the establishment of action correspondence identification and control and the access of action correspondence status information.

Reports Control Program (CRP-AA-07)

This procedure defines the actions and responsibilities for the evaluation of the usefulness of inter-Project Office reports. The procedure also describes the process for approval of new reports and establishes a Project Office Reports Directory.

Control of Project Office Procedures Manual (CRP-AA-11)

This procedure defines the actions and responsibilities for preparation, maintenance, and control of the Project Office Procedures Manual which contains copies of all Project Office procedures including those for quality assurance activities.

Controlled Documents (CRP-AA-14)

This procedure defines the responsibilities and actions of Project Office Divisions for controlled documents generated by Project participants.

Organization Plan and Functional Statements (CRP-DR-02)

This procedure defines the actions and responsibilities for maintenance of functional statements for Project Office organizational elements, delegations of authority and organization charts. The procedure also describes the process for securing approval for proposed changes.

Processing Principal Design Documents (CRP-EN-02)

This procedure defines the actions and responsibilities for conduct of internal reviews, reconciliation of comments and final action on Data Type I documents. This procedure also covers the internal review of Data Type 2, 3 and 4 documents submitted for information.

Processing Engineering Changes (CRP-EN-04)

This procedure defines the actions and responsibilities for review and approval of Class I Engineering Change Proposals (ECP) through the mechanics of a Configuration Control Board (CCB).

Configuration Control Board Actions (CRP-EN-05)

This procedure defines the actions and responsibilities of the Configuration Control Board in handling and disposing of Engineering Change Proposals (ECPs).

Preparation and Maintenance of the Project Level 1 Schedule (CRP-EN-09)

This procedure defines the actions and responsibilities for preparing and maintaining the Level 1 schedule. The procedure also describes the processes for monitoring, controlling, and documenting progress and changes.

Operations Division Review and Concurrence with Engineering Design Data (CRP-OP-02)

This procedure defines the actions and responsibilities for the review and concurrence for engineering data by the Project Office Operations Division. The procedure also describes the method used by the Operations Division for securing TVA Operations review and comments.

CRBRP Project Monthly Progress Report (CRP-PC-03)

This procedure defines the actions and responsibilities for the preparation, review, approval and distribution of the Project Monthly Progress Report.

TVA Purchases of CRBRP Items (CRP-PR-02)

This procedure defines the actions and responsibilities associated with CRBRP Procurement activity assigned to TVA. The procedure also describes the review and approval requirements.

Quality Assurance Program Management Review Meetings (CRP-QA-01)

This procedure defines the responsibilities and actions for planning, scheduling, conducting and the reporting of results of the bi-monthly, Project-wide quality assurance program management review meetings.

Activity Planning (CRP-QA-02)

This procedure defines the responsibilities and actions for the preparation of plans for the conduct of discrete quality assurance activities. The review of these activities in accordance with an activity plan is a part of the overall method for achieving quality.

Control of Nonconformances (CRP-QA-03)

This procedure defines the responsibilities and actions for the handling of nonconformances reported for items and services provided on the CRBRP Project. The procedure provides for the clearing or disposition of nonconformances observed.

Corrective Action Requests (CRP-QA-04)

This procedure defines the responsibilities and mechanics for initiating actions to correct improper work or conditions observed during the normal course of quality assurance activities.

Processing of Unusual Occurrence Reports (CRP-QA-05)

This procedure defines the responsibilities for evaluation of all Project related unusual occurrence reports. The procedure also details the actions of the Project Office Quality Assurance Division in terms of disposition.

Nonconformance, Unusual Occurrence and Corrective Action Analysis (CRP-QA-06)

This procedure defines the responsibilities for analysis of selected nonconformances, unusual occurrences, or corrective action reports for adequacy of corrective action proposed or taken. This procedure also details the processes for internal handling of these reports.

Quality Records (CRP-QA-07)

This procedure defines the actions and responsibilities for establishing, implementing, operating and maintaining a Project Office Quality Records Center. The procedure also details the criteria for selection of data documents for quality records retention.

Quality Assurance Program Progress and Status Review and Reporting (CRP-QA-08)

This procedure defines the responsibilities and actions for the review of the overall Quality Assurance Program on a monthly and semi-annual basis. These reviews result in the preparation of comprehensive program status reports for the Project Director's information and action as appropriate.

Quality Trend Analysis (CRP-QA-09)

This procedure defines the responsibilities for continuous monitoring of product and programmatic data to discern and identify quality trends. The procedure also outlines the process for initiating corrective action.

Quality Assurance Review and Approval of Engineering Documents (CRP-QA-10)

This procedure defines the responsibilities for review of engineering specifications, descriptions, drawings and change proposals for adequacy in terms of quality assurance requirements. The procedure also details the process of documenting the review findings.

Quality Assurance Review of Procurement Documents (CRP-QA-11)

This procedure defines the responsibilities for review of procurement documents for adequacy as related to quality assurance requirements. The procedure also details the process of documenting the review findings.

Review of Contractor Quality Assurance Plans and Procedures (CRP-QA-12)

This procedure defines the responsibilities for review of CRBRP participants quality assurance plans and procedures to determine their adequacy. The acceptance/approval process is also described.

Performance of Project Surveillance (CRP-QA-13)

This procedure defines the responsibilities for the planning, conduct and reporting of surveillance activities performed by the CRBRP Quality Assurance Division.

Inspection, Examination and Test (CRP-QA-16)

This procedure defines the responsibilities for the preparation for and performance of quality assurance inspections, examinations and tests during design development, procurement, construction, installation, start-up, operation, maintenance and modification of the CRBRP.

Measuring and Test Equipment Calibration and Control (CRP-QA-17)

This procedure defines the actions and responsibilities for the verification that measuring and test equipment used for inspections, examinations or tests are properly calibrated and controlled.

Quality Assurance Participation in Preparation of Approved Source Lists (CRP-QA-18)

This procedure defines the responsibilities for identifying qualified sources of supply for BOP items and services. The procedure also outlines the processes and actions necessary for the evaluation of a prospective supplier's quality assurance capabilities.

Administration of Quality Assurance Auditing (CRP-QA-19)

This procedure defines the responsibilities for the planning, conduct, follow-up, and close-out of quality assurance audits. This procedure also details the actions of the quality assurance audit administrator in documenting the audit activity.

Preparation, Maintenance and Control of Project Office Quality Assurance Manual (CRP-QA-20)

This procedure defines the actions and responsibilities for the preparation, distribution, maintenance and control of the CRBRP Quality Assurance Manual.

Conduct of Product Audits (CRP-QA-21)

This procedure defines the actions and responsibilities for the preparation, conduct and reporting of quality assurance product audits by the CRBRP Project Office. The procedure also details the actions of the audit team in the course of the evaluation of selected products for conformance to quality requirements.

Conduct of Programmatic Audits (CRP-QA-22)

This procedure defines the responsibilities for the preparation, conduct, and reporting of quality assurance programmatic audits by the CRBRP Project Office. The procedure also details the actions of the audit team in the course of the evaluation of programmatic practices for conformance to quality program requirements.

Preparation, Transfer, and Receipt of Project Office Quality Records (CRP-QA-23)

This procedure defines the responsibilities and actions to be executed by each Project Office Division in the preparation and transfer of quality records to the Quality Assurance Division. The procedure also defines the responsibilities and action of the Quality Assurance Division when receiving quality records from other Project Office Divisions.

Personnel Indoctrination (CRP-QA-24)

This procedure defines the responsibilities and actions to provide for the indoctrination of CRBRP Project Office personnel who carry out duties affecting the quality of CRBRP Plant structures, systems and components.

Administration of Personnel Certification and Records (CRP-QA-25)

This procedure defines the responsibilities for the administration of certification for Quality Assurance Division personnel directly involved in quality verification, testing, evaluation or audit activities. The procedure also details the actions associated with collection and maintenance of records pertaining to personnel certification.

Personnel Certification (CRP-QA-26)

This procedure defines the responsibilities and actions necessary to identify areas of quality importance for which qualification or certification of personnel are required. The procedure also details the actions for verifying the adequacy of personnel training programs, certification practices and documentation.

Unusual Occurrence Report Preparation and Disposition (CRP-QA-27)

This procedure defines the actions and responsibilities for documenting an unusual occurrence observed during the course of work on the CRBRP Project. The procedure also details the action related to evaluation of the reportability of the event to NRC as well as the channels for reporting to NRC.

AE PROCEDURES

Procedure Preparation (BRD-PC-1.5)

This procedure establishes the method for preparation, review, approval and updating of all Project procedures except those numbered "BRD-QA-xx."

Filing (BRD-PC-3.1)

This procedure establishes the methods and categories to be used in Project files; it contains provisions for a QA history file in compliance with Criterion VII of 10CFR50, Appendix B.

Distribution (BRD-PC-3.6)

This procedure establishes the method for maintaining distribution requirements of Project documents.

Indoctrination and Training (BRD-PC-7.1)

This procedure establishes the requirements for indoctrination of Project personnel in the goals, policies and procedures of the Project, training in the Project work methods and provides documentation of accomplishment of the procedure activities.

Prequalification of Bidders (BRD-E-2.2)

This procedure establishes the method of prequalification of prospective bidders for technical and financial capability. It also provides for generation of a prospective bidder's list using the above information and data accumulated via BRD-QA-1.11.

Technical Evaluation of Bids (BRD-E-2.3)

This procedure defines the methods to be used for engineering review and evaluation of bids.

Vendor/Contractor Documents (BRD-E-2.4)

This procedure provides the methods for receipt, logging, review, processing and return of vendor/contractor documents.

Vendor/Contractor Waiver Requests (BRD-E-2.5)

This procedure provides the method by which vendor/contractor waiver requests are received, evaluated, dispositioned and the vendor notified of the results.

Reporting of Defects and Non-Compliance (BRD-L-2.3)

This procedure establishes the method for review of defects or non-compliance as defined by 10CFR21 and significant deficiencies as defined by Paragraph 50.55(e) of 10CFR50.

Preparation, Control and Distribution of Quality Assurance Instructions (BRD-QA-1.2)

This procedure establishes the guidelines for preparation, issue, control and use of quality assurance instructions by QA personnel.

Preparation of QA Procedures (BRD-QA-1.3)

This procedure establishes the method for preparation and control of "QA" designated procedures.

Vendor Quality Assurance Prequalification Program (BRD-QA-1.11)

The procedure establishes the method of prequalifying a prospective bidder's quality assurance program for a bidder's list.

Vendor Quality Assurance Qualification Survey (BRD-QA-1.12)

This procedure establishes the method and criteria for conducting a preaward survey and evaluation of a prospective vendor's or subcontractor's quality assurance/quality control system.

Corrective Action Request (CAR) (BRD-QA-1.13)

This procedure establishes the methods and documentation used in requesting corrective/preventive actions via a system of graded requests.

Quality Assurance Review of Submittals (BRD-QA-1.16)

This procedure provides guidelines for a standard approach to quality assurance review of submitted documents.

Training and Certification of Quality Assurance Personnel (BRD-QA-1.18)

This procedure establishes the training and certification methods for quality assurance personnel who perform nondestructive examinations and inspection of materials, parts, structures or systems.

Procedure Writing Format (BRD-QA-1.19)

This procedure provides a guide to the standardized format to be used in writing procedures for the BRD.

Bid Review for Quality Requirements (BRD-QA-1.21)

This procedure provides guidelines for a standard approach for quality assurance review of bids.

Nonconformance Review Board (NRB) (BRD-QA-1.25)

This procedure provides guidelines for the composition, operation and generation of documentation by the Nonconformance Review Board. It provides for membership of the Authorized ASME Inspector when considering items under the jurisdiction of the ASME Code.

Deviation Reporting and Control (BRD-QA-1.1000)

This procedure establishes the methods and documentation required to report and disposition deviations.

Source Surveillance (BRD-QA-3.101)

This procedure establishes the scope, guidelines, responsibilities and control of source surveillance activities, from initial planning through release for shipment.

Project Surveillance (BRD-QA-3.1000)

This procedure provides a means and guidelines for examining the effectiveness of the Project Quality Assurance Program on a less formal basis than auditing.

Audits (BRD-QA-4.2)

This procedure establishes the guidelines for auditing of the Quality Assurance Program.

Evaluation of Prequalification Questionnaire (BRD-QA-1.11-1)

This instruction provides direction for evaluating a Q.A. Prequalification Questionnaire when considering the suitability of a vendor as an acceptable source.

Performance, Evaluation and Reporting of Preaward Surveys (BRD-QA-1.12-1)

This instruction defines the actions required in planning, performing and reporting the results of a preaward survey.

Review of Design/Document Submittals (BRD-QA-1.16-1)

This instruction provides the checklist that defines the minimum QA review of design documents and vendor submitted documents.

Preparation of Project Surveillance/Acceptance Checklists, Summary and Report Forms (BRD-QA-3.1000-1)

This instruction defines the requirements for a surveillance checklists, methods of summarizing the results and reporting them.

Quality Assurance Completion Record (BRD-QA-3.101-2)

This instruction provides for recording the condition of a component and its documentation immediately prior to shipment.

Preparation For and Performance of Surveillance (BRD-QA-3.101-3)

This instruction describes the methods and considerations addressed when preparing and performing surveillance activities.

Preparation and Issuance of Source Surveillance Reports (BRD-QA-3.101-4)

This instruction defines the requirements and provides the guidance for preparation and issuance of Source Surveillance Reports.

ATTACHMENT II

PROCEDURE RELEASE SCHEDULE

RESERVED

CLINCH RIVER BREEDER REACTOR PLANT
A DESCRIPTION OF THE CONSTRUCTOR
QUALITY ASSURANCE PROGRAM

TABLE OF CONTENTS

CLINCH RIVER BREEDER REACTOR PLANT
DESCRIPTION OF THE CONSTRUCTOR
QUALITY ASSURANCE PROGRAM

0.0 INTRODUCTION

Stone & Webster Engineering Corporation (SWEC) is the constructor for the Clinch River Breeder Reactor Plant (CRBRP) Project. In this capacity, SWEC is responsible for management and performance of those tasks associated with the overall construction effort. This includes the responsibility to plan, implement, and manage the Constructor portion of the CRBRP overall quality assurance program. This program will be applied to activities within SWEC's contractual scope of work that affect safety related structures, systems, and components and those important to safety (as defined in Section 3.2, 7.1, and 9.13).

The SWEC CRBRP Project Quality Assurance Program is based on the Stone & Webster Topical Report, SWSQAP 1--74A, Rev. C, "Standard Nuclear Quality Assurance Program." Although some organizational elements and responsibilities have been shifted for this project, all requirements contained in SWSQAP 1-74A, Rev. C, which are applicable to SWEC's scope of work will be implemented. By accomplishing this, the SWEC Quality Assurance (QA) Program for CRBRP complies with the applicable requirements of 10CRF50, Appendix B and RDT F2-2. The correlation of 10CRF50, Appendix B and RDT F2-2 is shown in Figure 17.1-3.

The changes that have been made in the quality assurance organizational structure from that shown on SWSQAP 1-74A, Rev. C, have been made to respond to project conditions where SWEC does not have responsibility for engineering or design, as well as requirements of the Owner for the establishment of a project quality assurance organization. As a result, the responsibilities for implementing some requirements contained in SWSQAP 1-74A, Rev. C, have been shifted within the organizational elements of the QA Department. These changes in organization and responsibility are described in the following paragraphs.

0.1 ORGANIZATION

0.1.1 Organizational Arrangement

The SWEC management organization, including quality assurance management for the CRBRP Project is shown in Figure 17F-1. The QA Department organization is shown in Figure 17F-2. Figure 17F-3 shows the Project Quality Assurance Organization. The changes in the quality assurance organizational structure from that presented in SWSQAP 1-74A, Rev. C, for the Constructor program of the CRBRP Project are:

A. Because SWEC has no engineering or design responsibility for CRBRP, these organizatinal elements are not represented.

B. The position of CRBRP Project QA Manager has been created. The CRBRP Project QA Manager, who will be located at the project site, has overall authority and responsibility for quality assurance function, both administrative and operational, on the project. The Project QA Manager receives quality assurance direction from, and reports to, the QA Department Manager in SWEC Headquarters. The QA Department Manager reports to the Vice President, QA, who reports to the SWEC Company President. CRBRP Project policy is received through the interface shown in Figure 17F-1 with the Project Managers and/or the Senior Project Manager.

C. The position of QA Program Administrator will not be established for the CRBRP Project. The Project QA Manager and the Project QA Staff will perform those functions normally assigned to the coordinator.

D. Field Quality control Division personnel at the site and Procurement Quality Assurance personnel in the SWEC District Offices will receive Project direction from the Project QA Manager. Corporate policy, corporate administration, and corporate resource support will remain with the parent Headquarter's divisions.

E. The Project QA Manager's Staff will be established to perform both quality assurance engineering and quality assurance management functions.

F. The Quality Assurance Cost and Auditing Division in SWEC Headquarters will retain responsibility for audits of the overall SWEC CRBRP Project QA Program. Additionally, the Project QA Manager will staff and conduct an audit activity to perform required audits of subcontractors, the S&W CRBRP FQC organization, and others as requested or directed by the Owner.

G. With respect to procurements, the site QA staff will be responsible for inspection planning normally performed by PQAD and for the evaluation of risk releases normally performed by Project Engineering. The field Procurement Department, in lieu of Project Engineering, will assume the responsibility for preparation of the Recommended Bidder's List.

H. Since the engineering and design activities for CRBRP are the responsibility of others, the assignment of dispositions to nonconforming items identified by SWEC and which require an Engineering disposition will be the function of others. To facilitate inter-organizational processing, the SWEC Nonconformance and Disposition (N&D) Report System (referenced in Section 15) has been modified. The new system, designated as the Field Deviation and Disposition Report (FDDR) provides a method of interfacing with all responsible parties in order to report, evaluate, and disposition nonconformances. PQA will continue to report shop nonconformances using the SWEC N&D system.

I. Additional Modifications to SWSQAP 1-74A, Revision C

1. Section 1, paragraph 1.3, revise to read - individuals or groups who audit, inspect, or otherwise provide acceptance verification of a quality activity (except for design or start-up operations) shall be from the quality assurance organization and shall not be the same individuals or groups responsible for performing the specific activity.

2. Section 1, add paragraph 1.4 as follows: The quality assurance organization shall be adequately staffed throughout the life of the project. This organization shall review the project scope, determine the personnel requirements to support quality assurance activities, and staff to provide required support. A Quality Assurance Representative shall participate in scheduling meetings and other day-to-day activities at the site and Headquarters as necessary to assure adequate qualified personnel, equipment, and procedures are available to perform quality activities in support of the engineering and construction schedule.

3. Section 1, paragraph 2.0 - In addition to the described tasks, the Constructon Department shall develop management systems and methods to implement the quality assurance program for Construction Department activities.

4. Section 1, paragraph 2.0 - In additon to the described tasks, the Purchasing Department shall develop management systems and methods to implement the quality assurance program for Purchasing Department activities.

5. Section 2, add paragraph 1.10 as follows: All procedures which are used to implement this quality assurance program shall be consistent with the commitments of this program. The QA Organization shall review and concur with these quality related procedures in accordance with Appendix VI. Documentation of the review and concurrence shall be maintained.

6. Section 2, add paragraph 1.11 as follows: The status and adequacy of the overall Quality Assurance Program, as described herein, shall be assessed on a annual basis by the Stone & Webster Internal Audit Division or other organization having no direct relationship to the SWEC Quality Assurance Organization. This annual assessment will be achieved by an evaluation of compliance to and adequacy of corporate commitments contained in SWSQAP 1-74A, implementing measures including the project related assessments committed to in Section 18 of this document. Reports of this assessment and recommendation shall be submitted to the Office of the Chairman and President of SWEC and the SWEC Vice President and Manager of Quality Assurance.

7. Section 2, paragraph 1.8, revise to read: Indoctrination, training, and qualification programs shall be established and implemented as appropriate, such that:

1.8.1 Personnel receive indoctrination and training to familiarize them with the procedures and systems developed to govern and support quality related and quality assurance activities, including tests, inspections, examinations, and audits.

1.8.2 Formal training programs shall be documented, including objectives, content of the program, attendees, and date of attendance.

1.8.3 Personnel performing quality assurance functions shall be qualified, certified, and re-certified as required by applicable codes and standards.

1.8.4 Certificates of qualification show the basis for qualification, including testing or proficiency testing when applicable, and the specific functions personnel are qualified to perform.

1.8.5 The training program complies with the Regulatory Position in Regulatory Guide 1.58, with alternatives as noted in Appendix VII.

8. Section 4, paragraph 2.0 - In addition to the described tasks, the Construction Department is responsible for the preparation and processing of Purchase Requisitions for permanent plant equipment, materials, and services as assigned by the Owner or as necessary to support construction phase activities.

9. Section 4, paragraph 2.1, revise as follows: First sentence - change to read: Quality Systems Division, or other personnel authorized in writing by the Chief Engineer, Quality Systems Division, shall be responsible....

   Second sentence - Review and approval shall verify that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and sufficient information exists pertaining to codes....

10. Section 5, paragraph 2.0 - In addition to the described tasks, the Construction Department shall prepare and publish appropriate Construction Department procedures which govern the performance of Construction Department activities affecting quality.

11. Section 5, paragraph 2.0 - In addition to the described tasks, the Project Management Department shall prepare a Project manual that provides overall direction to Project personnel, reference applicable detailed procedures and instructions, and contain Project unique procedures and instructions.

12. Section 5, add paragraph 2.2.3 as follows: In addition to the described tasks, the Project Quality Assurance Organization shall prepare and publish procedures based on Quality Systems Division procedures without degradation which shall govern the performance of certain Project quality activities.

13. Section 6, paragraph 2.0 - In addition to the described tasks, the Construction Department shall prepare and issue procedures establishing a document control system for documents received and distributed for use at the site which prescribe quality assurance activities.

14. Section 6, paragraph 3.1, revise to read - Applicable procedures in Section 5 of the Quality Standards Manual, Engineering Assurance Manual, and the Quality Assurance Directives Manual establish the requirements to maintain master indexes of instructions, procedures, drawings, and procurement documents and to publish updated indexes in a scheduled manner. For example, the Quality Standard entitled "Quality Standards Procedural System" states as follows:

> "7.1.7A. Applicability - The applicability of generic QS's to a major project shall be established during the procedure review cycle and documented in the Table of Contents and Project Applicability Matrix. Actual usage in part or whole will depend upon other project documents which establish the scope of work to be done by Stone & Webster. This matrix shall be reviewed and updated periodically depending on activity, with an annual update as a minimum. Major projects shall individually issue a Table of Contents for their Project QS Manual which addresses every master generic and project model QS applicable to the Project. The Table of Contents shall state adoption, projectization, and, if not used, shall so state. Further, if not used and a substitute project procedure is used (not a QS), it shall be noted in this Table of Contents to ensure procedural coverage of all program commitments."

15. Section 7, add new paragraph 1.7.1 as follows: Material procured with a Certified of Conformance as documentation of quality shall be receipt inspected periodically by Field Quality Control to verify compliance with procurement documents.

    Renumber present 1.7.1 through 1.7.4.

16. Section 7, paragraph 1.7.1 (formerly 1.7.1), revise to read - Receipt inspection status shall be documented and shall be identified by markings, tags, or other appropriate means.

17. Section 7, paragraph 3.2 - In the third line, change "Test, Inspection, and Documentation Section of procurement specifications...." to read "procurement documents and inspection plans...."

18. Section 8, paragraph 2.0 - In addition to described tasks, the Construction Department shall receive items at the site, verify proper quantity, item type, and lack of shipping/handling damage, and notify FQC for receipt inspection. Items which are awaiting FQC receipt inspection for a period exceeding one working day shall be tagged with a Product Hold Tag by Construction and segregated when practical, pending FQC receipt inspection.

19. Section 10, paragraph 1.2, revise first sentence to read - Inspection requirements shall be translated into inspection procedures, inspection plans, and inspection reports to provide documentation of the inspection work required to ensure the specified quality.

20. Section 10, paragraph 1.3, revise to read - Sampling techniques may be utilized for inspecting a group of homogeneous items. If sampling is used to verify the acceptability of items, the sampling plan shall be based on a recognized standard sampling plan (MIL-STD-105D for attribute sampling, MIL-STD-414 for variables sampling) or other nationally recognized and accepted technique. The method utilized and conclusions obtained from sampling shall be documented to assure correct interpretation of the plan and the results. Quality Systems Division and Client approval of sampling plans for Category I items is required when the method is outside the scope of approved procedures or accepted techniques as described above.

21. Section 10, paragraph 2.0 - In addition to the described tasks, the Construction Department is responsible for notification to FQC when work approaches HOLD points.

22. Section 10, paragraph 3.2.4, revise to read - <u>A Description of Method of Inspections with Equipment Requirements and Accuracy Criteria</u> - Delineated in the appropriate Quality Assurance Department document, i.e., Quality Assurance Directive, Inspection Plan, etc.

23. Section 12, paragraph 1.2, revise to read - The control program shall include the following and shall be implemented in accordance with approved procedures.

24. Section 12, paragraph 1.2.1, revise to read - Positive identification of the equipment and its calibration status including the due date of the next calibration.

25. Section 12, paragraph 1.2.2, revise to read - A frequency of calibration schedule for types of equipment based on required accuracy, purpose, recognized industry standards, manufacturers' recommendations, usage factors, stability characteristics, and other conditions affecting the measurement.

26. Section 12, paragraph 1.2.3, revise to read - Written procedures describing the calibration control system. Standards traceable to national standards shall be used; if national standards do not exist, the basis for calibration shall be documented. Calibration standards used shall be calibrated, where possible, using standards of a greater accuracy, or when not possible, the basis of acceptance of the calibration shall be documented. Calibration of equipment shall be against standards that have an accuracy that assures the equipment being calibrated will be within required tolerance. When possible, calibration standards used shall have an accuracy at least four times that of the equipment being calibrated. When this is not possible or feasible, standards shall have a verifiable accuracy which will assure that the calibrated equipment will be within required tolerances.

27. Section 15, paragraph 1.4 - Add "and 10CFR21" to end of sentence.

28. Section 15, add new paragraph 1.5 as follows: Nonconforming items shall either be corrected, or resolved as not having an adverse impact upon the test or test results, prior to the initiation of the preoperational test program on the item.

29. Section 18, paragraph 2.1, revise to read - The Vice President, Quality Assurance, shall review the implementation of each project's quality assurance program for compliance with the Preliminary Safety Analysis Report and Appendix B to 10CFR50. This shall consist of reviews of quality trend data, audit reports, and specific project reports, and verification of implementation of effective corrective action. At least annually, a formal program audit of the specific project quality assurance program shall be performed and documented to evaluate program effectiveness and determine whether the Preliminary Safety Analysis Report requirements are properly reflected in the various quality assurance manuals and are being, or are capable of being, fulfilled. These audits shall include pertinent external project-related activities such as, but not limited to, those performed by Construction, Purchasing, and Project Management Departments, and Engineering Assurance Division of the Engineering Department. Appropriate corrective action shall be identified. Copies of the program audit, and identified corrective action shall be submitted to responsible management to implement corrective action. Corrective action shall be tracked to completion.

30. Section 18, paragraph 3.2.1, line 2: Replace "total" with "each project."

31. Section 18, paragraph 3.2.2: Replace second sentence with: This evaluation shall be conducted by QACAD and reported to the Manager and Vice President of Quality Assurance. This program feature shall be assessed annually by the selected independent auditors identified in Item 0.1.1.1.6 above.

32. Section 19, paragraph 1.8, revise to read - Owner requirements and/or additions to the Project QA Program Manual shall be entered into the Quality Assurance Program Index (Division A of the Program Manual) or in Part 4.0, "Client Considerations," within each section and shall not be considered as a program revision.

33. Section 19, paragraph 1.9, add the following: In addition, any changes to the quality assurance program description which is included with the Safety Analysis Report, and have been previously approved by the NRC, will be submitted to the licensee for the purpose of obtaining NRC approval prior to implementation. The licensee shall be notified of organizational changes in the quality assurance organization within 30 days after the announcement for the further notification of these changes to the NRC.

34. Appendix II - Add the qualification requirements of the Project QA Manager for the CRBRP Project as follows: A minimum of ten years in quality assurance and related fields including manufacturing, construction, and/or installation activities. At least two years of this experience shall be associated with the nuclear field in either field or Headquarters project quality assurance assignments. He must have a Bachelor of Science or Arts degree.

35. Appendix VI - Add Project QA Manager to the approvals for: Project QA Program Description, Quality Standards (other than QS 5.1), Quality Assurance Directives, and Quality Control Instructions.

36. Appendix VII, revise Item D (Regulatory Guide 1.58) as follows:

Regulatory Guide 1.58, Revision 1, dated September 1, 1980, (ANSI N45.2.6 - 1978), "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel."

SWEC's QA program requirements commit to comply with this Regulatory Guide subject to the following alternatives:

1. ANSI N45.2.6 - 1978, Paragraph 2.4, "Written Certification of Initial Capability"

   SWEC Position: Initial capability will be determined by an evaluation of the candidate's education and experience or by testing. If the candidate fails to meet the criteria established in Paragraph 3.5, subject to our alternatives stated below in Item 3, SWEC will evaluate the candidate by testing. Testing to demonstrate proficiency will be accomplished by a practical demonstration, oral or written examination, or by any suitable combination of the three. In all cases, the basis for the qualification and the results will be documented in an auditable manner and retained in the candidate's qualification file. Evaluation by testing may be optionally exercised at anytime in lieu of verified education and experience.

2. ANSI N45.2.6 - 1978, Paragraph 2.4, "Written Certificate of Qualification"

   SWEC Position: For purposes of certification, SWEC will use the following disciplines on certificates of qualification to identify activities certified to perform:

   - Mechanical - (includes piping and instrumentation)
   - Electrical - (includes controls)
   - Civil - (includes concrete, structural and soils)
   - Special Processes (except NDT - see below) - (includes welding, painting, chemical, and cadwelding)

- o <u>Quality</u> - (includes supervising personnel who review or administer inspections, examinations, or tests over several disciplines, as well as multi-disciplines, such as receiving inspection, procurement quality assurance, documentation, etc.)

- o <u>NDT Disciplines</u> - (as delineated in SNT-TC-1A-1975)

  Certification will be accomplished either by (1) education, experience and training, or (2) testing. The method used will be shown on the certificates. Results of testing and records of education, experience, and training will be maintained in the candidates qualification file.

3. <u>ANSI N45.2.6 - 1978, Paragraph 3.5, "Education and Experience - Recommendation"</u>

<u>3.5.1 - Level I</u>

1. Same as Standard

2. High School/General Education Development equivalent plus six months...or

3. Four year college graduation, plus one month of related experience or equivalent inspection, examination, or testing activities.

<u>3.5.2 - Level II</u>

1. One year of satisfactory performance as Level I or five years related experience in the corresponding inspection, examination, or test category or class, or

2. High School/General Education Development equivalent plus three years...or

3. Same as Standard

4. Same as Standard

<u>3.5.3 - Level III</u>

1. Six years of satisfactory performance as a Level II or 15 years of related experience in the corresponding inspection, examination, or test category or class...or

2. High School Graduation/General Education Development equivalent plus ten years...or

3. Same as Standard

4. Same as Standard

37. Appendix VII, add to Part 1:

L. Regulatory Guide 1.144, Rev. 1, dated September 1980 (ANSI N45.2.12 - 1977), "Auditing of Quality Assurance Programs for Nuclear Power Plants" - commit to comply with Guide, subject to the following alternative:

Pre-audit and post-audit conferences are normally held as required by Sections 4.3.1 and 4.3.3 of ANSI N45.2.12 - 1977. In certain circumstances, when audits are of a limited scope, a documented, telephone conversation may be held in lieu of a face to face meeting.

M. Regulatory Guide 1.146, dated August 1980 (ANSI/ASME N45.2.23 - 1978), "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" - commit to comply with Guides.

0.1.2 Responsibility and Authority

The shift of quality assurance responsibilities to support the revised organizational structure is as follows:

A. Because SWEC has no engineering and design responsibilities, these functions will be performed by others designated by the Owner as described in other sections of this PSAR. Additionally, as-built configuration information is provided by SWEC to others as designated by the Owner for incorporation in to the Project Baseline Documentation which will ultimately reflect the as-built configuration of the overall plant.

B. The Project QA Manager is responsible for performing the quality assurance program management and administrative functions for the project quality assurance organization, as delegated by the QA Department Manager. In addition, the Project QA Manager is responsible for those tasks normally assigned to a QA Program Administrator. The QA Department Manager, located in Boston Headquarters, will provide quality assurance policy and guidance, the interface with corporate management, and access to other QA Headquarters divisions. In this position, the Project QA Manager has the organizational freedom and authority to identify quality problems, initiate, recommend, or provide solutions through designated channels, and verify implementation of corrective action. The Project QA Manager is also responsible for establishing necessary interfaces with other project participants on both informal and formal basis as designated by the Owner. The Project QA Manager shall also participate in the long and short range planning and scheduling activities of the Project to assure that based on Project goals and schedules, the overall quality assurance function has adequate resources to support all phases of Project work efficiently.

C. The Project QA Staff will execute many of the Project QA Manager's assigned tasks for the Constructor Program.

D. The Quality Assurance Cost and Auditing Division in SWEC Headquarters will retain responsibility for audits of the overall SWEC CRBRP Project QA Program. Additionally, the Project QA Manager will staff and conduct an audit activity to perform required audits of subcontractors, the SWEC CRBRP FQC organization, and others as requested or directed by the Owner.

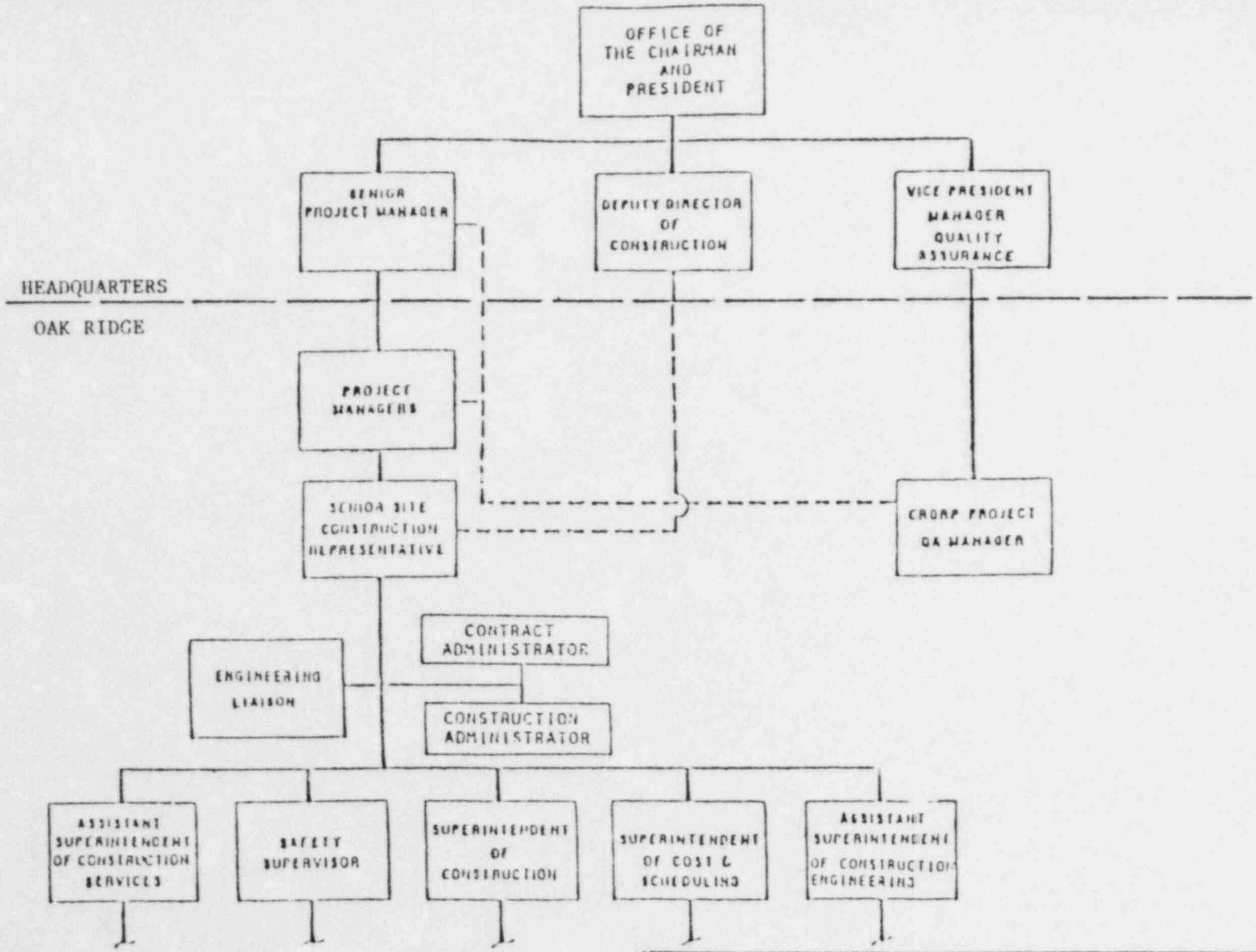
0.1.3 Qualification Requirements of the Project QA Manager

The qualification requirements of the Project QA Manager are described in Section 1.4.4.6 and paragraph 0.1.1.1.34, above.

0.2 PROGRAM

The Constructor Quality Assurance Program is a major poriton of the overall Project Quality Assurance Program. The scope of the program and the type of project participation covered by the program are shown in Figure 17F-4.

The major elements of the Constructor Program are shown in Figure 17F-5. SWEC has been assigned execution responsibility for the full scope of the Constructor program except the area of preoperational testing and start-up activities. Responsibility for execution of activities related to those areas has been retained by the Owner.



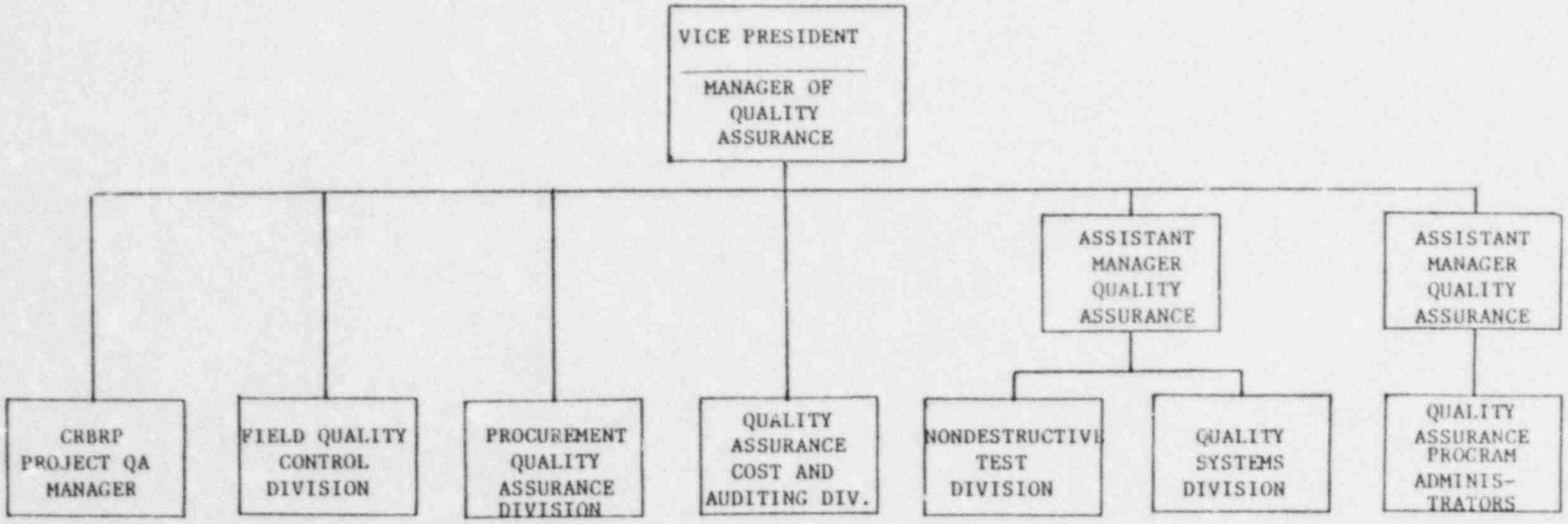
LEGEND

——— RESPONSIBILITY

- - - - COMMUNICATION & LIAISON

PROJECT ORGANIZATION FOR QUALITY ASSURANCE
CLINCH RIVER BREEDER REACTOR PLANT PROJECT
STONE & WEBSTER ENGINEERING CORPORATION

Figure 17F-1



NOTE: PROJECT DIRECTION IS PROVIDED BY THE CRBRP PROJECT QA MANAGER TO OTHER ORGANIZATIONAL UNITS OF THE QA DEPARTMENT WHICH ARE PERFORMING ACTIVITIES IN SUPPORT OF THE PROJECT.

QUALITY ASSURANCE DEPARTMENT ORGANIZATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT
STONE & WEBSTER ENGINEERING CORPORATION

Figure 17F-2

Amend. 70
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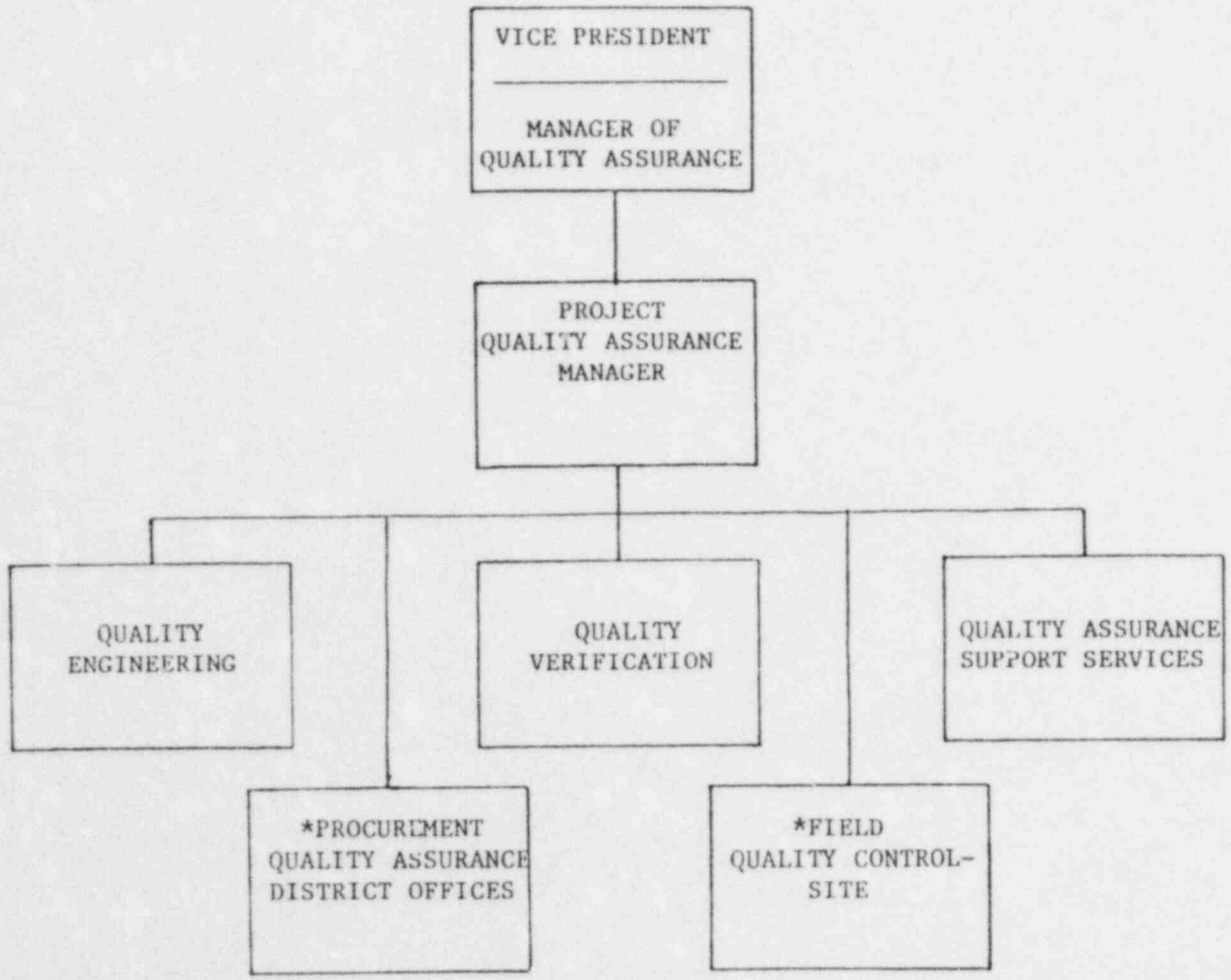

*NOTE: PROCUREMENT QUALITY ASSURANCE AND FIELD QUALITY CONTROL PERFORM VERIFICATION ACTIVITIES AS AN INTEGRAL PART OF THE PROJECT QUALITY ASSURANCE ORGANIZATION AND RECEIVE PROJECT DIRECTION FROM THE PROJECT QA MANAGER. CORPORATE ADMINISTRATION, CORPORATE POLICY, AND CORPORATE RESOURCE SUPPORT ARE RECEIVED FROM THEIR PARENT DIVISIONS IN BOSTON HEADQUARTERS.

PROJECT QUALITY ASSURANCE ORGANIZATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT
STONE & WEBSTER ENGINEERING CORPORATION

Figure 17F-3

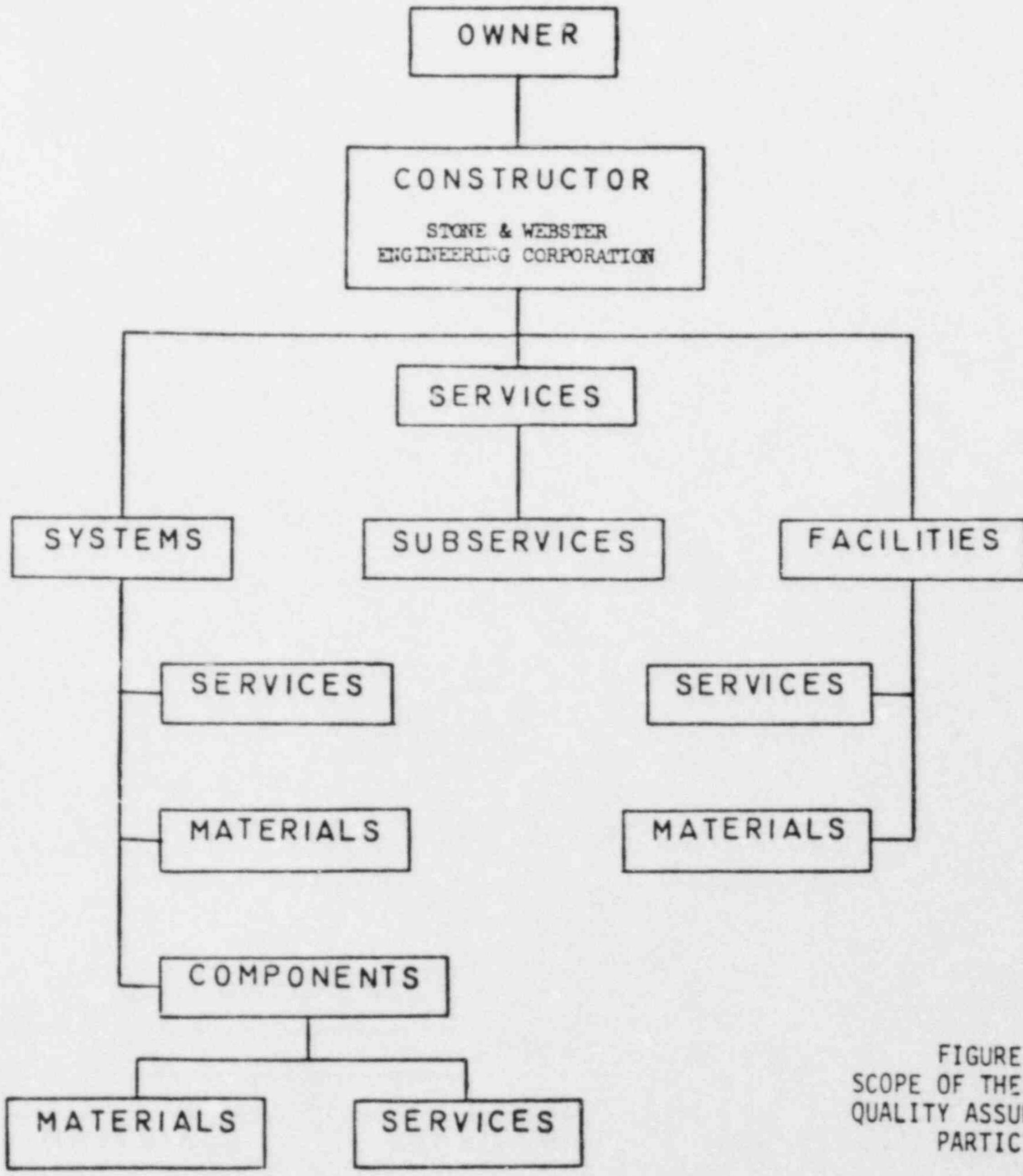

FIGURE 17F-4
SCOPE OF THE CONSTRUCTOR
QUALITY ASSURANCE PROGRAM
PARTICIPATION

CONSTRUCTOR PROGRAM ACTIVITIES

PROGRAM MANAGEMENT

- QUALITY ASSURANCE PROGRAM
  1. PLANNING
  2. QUALITY ASSURANCE PROGRAM INDEX
- ORGANIZATION
  1. RESPONSIBILITY AND AUTHORITY
  2. TRAINING AND INDOCTRINATION
  3. PERSONNEL QUALIFICATION
- DOCUMENTATION
  1. POLICIES AND PROCEDURES
  2. QUALITY RECORDS
  3. QUALITY STATUS REPORTS
- AUDITS AND REVIEWS
  1. QUALITY AUDITS
  2. MANAGEMENT REVIEWS
- CORRECTIVE ACTION
- ENGINEERING HOLDS
- UNUSUAL OCCURRENCE REPORTS

PROCUREMENT

PROCUREMENT PLANNING

PROCUREMENT REQUIREMENTS

PROCUREMENT DOCUMENT REVIEW

EVALUATION AND SELECTION OF PROCUREMENT SOURCES
1. GENERAL REQUIREMENTS
2. ACCEPTABLE SOURCE LISTS
3. PREAWARD EVALUATION
4. INTERCHANGE OF SOURCE CAPABILITY INFORMATION

CONTROL OF CONFIGURATION
1. CONTRACT CHANGE CONTROL
2. AS BUILT VERIFICATION

EQUIPMENT CALIBRATION AND STANDARDS

SOURCE SURVEILLANCE AND INSPECTION

RECEIVING INSPECTION
1. PLANNING AND INSPECTION
2. DOCUMENTATION
3. DISPOSITION OF RECEIVED ITEMS

CONTROL OF NONCONFORMING ITEMS

CONTROL OF RECEIVED ITEMS

MANUFACTURING, FABRICATION, AND ASSEMBLY

PLANNING

INSPECTION AND TEST PLAN

MATERIAL IDENTIFICATION AND CONTROL

CONTROL OF PROCESSES
1. FABRICATION AND ASSEMBLY PROCESSES
2. PROCESS QUALIFICATION
3. NONDESTRUCTIVE EXAMINATION
4. CLEANING

INSPECTION AND TESTS
1. GENERAL REQUIREMENTS
2. PROCEDURES
3. COMPLETED ITEM INSPECTION AND TEST
4. INSPECTION STATUS INDICATION
5. CERTIFICATION

DOCUMENT CONTROL

EQUIPMENT CALIBRATION AND STANDARDS
1. EQUIPMENT EVALUATION
2. CONTROL OF INSPECTION MEASURING AND TEST EQUIPMENT
3. CALIBRATION STANDARDS
4. DISCREPANT EQUIPMENT

STATISTICAL QUALITY CONTROL AND ANALYSIS

CONTROL OF NONCONFORMANCE ITEMS

CORRECTIVE ACTION

HANDLING, PRESERVATION, PACKAGING STORAGE AND SHIPPING
1. HANDLING
2. PRESERVATION, PACKAGING AND STORAGE
3. SHIPPING

CONSTRUCTION AND INSTALLATION

ORGANIZATION

CONTRACT

CONSTRUCTION PLANNING

CONTROL OF CONFIGURATION

CONSTRUCTION PROCUREMENT

CONSTRUCTION AND INSTALLATION CONTROL
1. MATERIAL HANDLING AND CLEANING CONTROL
2. SPECIAL PROCESS CONTROL
3. TRAINING AND CERTIFICATION
4. INTERFACE CONTROL
5. EQUIPMENT CALIBRATION AND STANDARDS
6. CONTROL OF NONCONFORMING ITEMS
7. STATISTICAL QUALITY CONTROL AND ANALYSIS

INSPECTION
1. SOURCE INSPECTION
2. RECEIVING INSPECTION
3. SITE INSPECTION
4. INSTALLATION

TESTING, OPERATION AND MAINTENANCE
1. START UP OF EQUIPMENT AND SYSTEMS
2. PREOPERATION TESTING
3. OPERATION AND MAINTENANCE

Figure 17F-5 Major Elements of the Constructor Program

For the purpose of assessing the potential magnitude of a release of radioactive species from the cold trap, a rupture of the crystallizer tank was assumed followed by a sodium fire. The radioactive inventory in the cold trap was assumed to be that resulting from 15 years operation with 0.5 percent failed fuel. Leakage from the reactor containment building (RCB) to the confinement annulus was based upon a very conservative estimate of the RCB overpressure. The mitigating effect that the confinement annulus has on leakage to the atmosphere was neglected. Resulting irradiation doses were calculated for the site boundary location and a number of downwind locations. Results from this analysis (provided in Section 15.7.2.7 of the PSAR) indicated that large margins existed between the potential doses and the applicable guideline values. It was concluded that a malfunction of the cold trap system would not result in a violation of the 10CFR100 and equivalent guidelines previously outlined.

Radwaste System

The liquid radwaste system utilizes components such as evaporators, demineralizers and filters whose performance has been demonstrated in LWR's. Operating procedures and tests will assure that the system is performing as designed and within technical specifications.

Analysis has been performed for a postulated failure of a tank containing the largest inventory of radioactivity in the liquid radwaste system, i.e., the radwaste collection tank. This analysis is provided in Section 15.7.2.5 of the PSAR. An analysis of the effects and consequences of the event has been performed assuming no credit for the cell floor drains or operator actions. Postulated gaseous and liquid releases associated with this tank failure have been shown to be well within specified limits.

The gaseous radwaste system utilizes components such as cryostills and charcoal absorber beds which process and purify reactor cover gas and, as necessary, gas from inerted cells. Selected tests are to be performed during scheduled plant outage periods to ensure that components are performing consistently within specifications.

Analyses have been performed for postulated component failures for equipment which could contain significant inventories of gaseous radwaste. Design features for the CRBRP include permissible leak rate containment specifications and testing provisions, as necessary, to achieve off-site consequences in compliance with Federal guidelines. Preliminary analyses provided in Section 15.7.2.4 of the PSAR indicate that off-site consequences of postulated events for the gaseous radwaste system will be in conformance with specified limits.

Primary Coolant

The primary coolant will become radioactive from activation of the sodium and from release of radioisotopes from fuel pins in the core. A program has been established to assure the integrity of the primary coolant boundary (see Section 1.6 of the PSAR, Reference 2). A sodium leak detection system is provided to detect small leaks so that operator action can be taken to limit leakage. In addition, inert atmosphere, cell liners and the containment isolation system limit the consequences of postulated leaks to well below the specified limits (see Section 15.6.1.4 of the PSAR).

Reactor Core

Fuel located within the core represents a major inventory of radioactive species. Prevention of the release of radioactive species from the core must be accomplished by means of the reliable operation of the appropriate safety related systems. This reasoning led to the emphasis in the Reliability Program being placed on those systems, the malfunctions of which could lead to a loss of coolable core geometry. Loss of coolable core geometry is believed to occur following cladding melting. For the purposes of the Reliability Program evaluation it has been conservatively assumed that loss of coolable geometry will occur at the onset of coolant boiling.

Postulated events which could lead to the loss of coolable core geometry can be divided into three categories. These are (a) events which result from the failure to shutdown power generation within the core when required, (b) events which result in a failure to remove the residual heat from the core in its shutdown condition and (c) fuel failure propagation. The systems designed to perform functions which would prevent these failures are the Reactor Shutdown Systems (RSS) and the Shutdown Heat Removal Systems (SHRS). The focus of the Reliability Program activities is therefore on reliability enhancement and verification of these systems (RSS and SHRS). Interfacing equipment and systems are included in the scope of the program when their malfunction could result in a safety related malfunction of either the RSS or the SHRS. Fuel failure propagation evaluations presented in Section 15.4 of the PSAR eliminate consideration of that failure mode as a potential initiator of significant radiological release.

The RSS is comprised of a primary shutdown system and a secondary shutdown system. As a safeguard against common cause failures, these systems differ substantially both in design and mode of operation of electronic and mechanical components. Major segments of the Reliability Program are devoted to the primary and secondary shutdown systems.

Shutdown heat removal capability is provided via four redundant heat removal paths. Three of these paths utilize the three heat transport loops. The normal short term heat sink for these loops during reactor shutdown is the turbine generator condenser. In the event of failure of this heat sink, a backup heat sink is provided. The backup heat sink utilizes steam release to the atmosphere and an auxiliary feedwater system to provide an initial high capacity heat sink coupled with protected air cooled condenser heat exchangers for long term heat rejection. The protected air cooled condenser heat exchangers are also used during normal long term shutdown.

The fourth shutdown heat removal path is via the Direct Heat Removal Service (DHRS). The DHRS makes use of the sodium overflow-makeup system to extract hot sodium from the reactor vessel and return cold sodium. Circulation of sodium through the core is achieved by means of forced circulation in the primary loops. Heat extraction from the sodium overflow-makeup loop is achieved via a heat exchanger located in the sodium makeup line. This heat exchanger is coupled to the EVST NaK cooling loops. Heat sink capability in this system is supplied by the EVST air cooled heat exchangers.

C.1.3 Program Design

The design of the CRBRP Reliability Program has been dictated by a number of key requirements and constraints. Some of these requirements and constraints would be common to any Reliability Program while others are unique to the CRBRP. The following paragraphs (a) identify some of the more important factors considered in the design of the program and (b) indicate how these factors influenced the selection of program features.

C.1.3.1 Design Integration

A basic ground rule set prior to the initiation of design work on the Reliability Program was that the primary objective for the program would be reliability enhancement in the RSS and SHRS. The intent of this ground rule was to ensure a direct integration of the Reliability Program and the component/system design activities. The most readily visible program feature stemming from this integration are the component level qualitative reliability assessments and the system level Reliability Design Support Documents. These documents are produced by the appropriate Design Engineering organization for items listed on the Reliability Related Components List (defined in Section C.1.4). The document is subject to approval by the Reliability organization. The requirement for the Reliability Design Support Document has the effect of ensuring direct Reliability Engineering involvement in the design process.

Reliability verification was set as a very important second objective. Reliability verification is achieved by a combination of component and system feature tests run under both design conditions and overload conditions coupled with qualitative reliability analyses at both the component and system levels and quantitative analysis at the RSS and SHRS levels.

C.1.3.2 Qualitative Reliability Analysis

The initial step in the qualitative reliability analysis is a preliminary total system Failure Mode and Effects Analysis (FMEA) for both the RSS and SHRS. Results from this analysis are used as a means of providing an initial identification of those system features having a significant impact on the overall system reliability.

Refinement of the system level FMEA is then achieved by rebuilding it from the component level up. The basic building block used in this process is the component level FMEA produced as a part of the component reliability qualitative assessments. This procedure for refining the system level FMEA was adopted as a means of assuring that the technology used in the design of many CRBRP components is reflected in an accurate and balanced manner in the reliability evaluations. Examples of this new technology are the high temperature design rules used in the design of outlet plenum and hot leg components and the irradiation effects (swelling, creep and ductility loss) technology used in the design of core components. Application of this technology is handled by specialists who are experts in their particular field. By requiring that the system level FMEA be reconstructed using FMEA's produced at the component level, the program assures that all necessary specialists, especially those who are experts in these specialized areas, are involved in the production of the FMEA building blocks. This process minimizes the potential for failure mode omission in areas of specialist knowledge.

At the component level, the failure mode criticality and probability ratings (defined in Section C.2.1) are related only to that portion of the RSS or SHRS directly impacted by the failure. The component designer does not make final judgments relative to the total system response to the failure of his component. This restriction is imposed to assure that component level failure mode impacts are not distorted by judgments based on an inadequate understanding of the system design.

The FMEA's contained in the component level qualitative reliability assessments are used by the system designers and reliability engineers to build system level qualitative assessments. In using the component level FMEA's to assess the system level effects, the system designer working with the reliability engineer reviews each component level failure mode and may modify its criticality and probability ratings based upon his knowledge of the total system response to the postulated failure. Typical of the factors considered in this reevaluation are system redundancy and the potential for common cause failures. The system level qualitative assessment is continuously updated as the component level input data are updated. The component level data are updated as the component proceeds through the various stages of design, fabrication, etc. System level FMEA's are used to guide the application of Reliability Program resources.

In performing the system level qualitative reliability analysis, special attention is paid to any failure mode or initiating event which has the potential to produce concurrent failure of more than one element of a system or more than one system. Failures in this category are termed common cause failures. Some of the measures adopted in the Reliability Program to provide protection against this type of failure include (a) imposing a requirement for diversity in the design of the RSS and SHRS, (b) requiring that essential safety related systems be redundant and have excess capacity and (c) performing evaluations and recommending corrective action to mitigate the consequences of any feature of a system or interfacing component which has potential susceptibility to common cause failures.

C.1.3.3 Quantitative Reliability Analysis

Quantitative reliability analysis plays an important role in the CRBRP Reliability Program. Its primary uses are (a) as a tool for the evaluation of systems (b) as a means for evaluating random independent failure modes, (c) as a decision aid for selecting between alternative designs, (d) as a guide for the design of the testing and analysis program and (e) as the basis for sensitivity studies to evaluate the range of unreliability due to uncertainties. As part of these evaluations, a top level system failure probability is calculated.

It is important to note that all the primary uses for the numerical reliability analyses are as aids in decision making. The analyses are not intended to demonstrate compliance with a top level system failure rate. The decision not to set a numerical failure probability requirement for the RSS and SHRS is a reflection of the current developmental nature of numerical reliability analysis in the field of nuclear safety systems.

An update of the reference (3) Preliminary Reliability Prediction for CRBRP SHRS will be available in January 1983. This will be the final SHRS reliability assessment update based on failure state block diagram modeling. Subsequent quantitative assessments will be based on the SHRS fault tree model being constructed for the CRBRP Probabilistic Risk Assessment. The primary reason for basing future reliability program numerical predictions and studies on the PRA fault tree model is the potential for additional level of resolution being modeled for SHRS and the supporting systems. This level of modeling will allow more extensive studies of potential for supporting systems interactions.

C.1.3.4 Test Program Rationale

Major objectives for the test program are (a) an identification to the appropriate design group of potential failure points in the design of the components/systems, (b) a deterministic evaluation of component/system performance margins as defined by the difference between the design operating envelope and the success envelope as defined by the test and (c) where possible to generate sufficient statistical data to be able to make meaningful probabilistic predictions of the component/system performance.

Tests that support the Reliability Program can be placed in one of three categories. These are: (a) component level or system feature tests, (b) system level tests and (c) materials tests. The tests are chosen on the basis of the criticality of the failure modes identified by the FMEA.

A significant segment of the test program is devoted to component tests and system feature tests. Tests performed at this level make possible the investigation of a wide range of conditions including overload conditions up to the point where failure may occur. A major objective for these tests is the identification of any areas of the design where reliability enhancement is desirable or necessary. The tests also result in the definition of the multi-dimensional success envelope for the component or system feature. The boundaries of this envelope can be compared with the boundaries of the design envelope for the component or system feature. This comparison provides a deterministic measure of the performance margin inherent in the component/system feature design.

System level tests are specified as appropriate to evaluate wear related phenomena, identify any failure modes which are real-time dependent (dormant failures), identify failure modes related to manufacturing variations, evaluate the effects of maintenance and operating procedures and identify failure modes associated with interface features not included in the component/system feature tests. Particular emphasis in the system level tests is placed on providing as exact a simulation as possible of the actual reactor operating environment (e.g., large sodium loops are used to provide a dynamic sodium environment in the case of the mechanical control rod system tests). Accelerated life system level tests are run to beyond the design life for the critical system components in order to confirm that the system design life does not lie close to a wear dependent failure boundary. Output from the system level tests provides a deterministic confirmation of the margins between the system design and success envelopes. The schedule for the system level tests necessarily lags that for the component and system feature tests. Any system level design problem uncovered in these tests, however, can still be corrected prior to operation of the plant equipment.

C.1.4 Program Implementation Procedures

Top level Reliability Program requirements for the RSS and SHRS in CRBRP are defined in the overall plant design requirements documentation. These requirements are interpreted by the Lead Reactor Manufacturer (LRM) in consultation with the CRBRP Project Office (PO) and then placed as mandatory requirements on the Reactor Manufacturers (RM's) responsible for portions of the RSS and SHRS. The LRM retains responsibility for overall coordination of the Reliability Program activities within the RM organizations. A key element of the administrative procedures set up by the LRM to assure the correct implementation of the Reliability Program activities within the RM organizations is the RM Reliability Engineering/Design Engineering Interface Definition Chart. When it is determined that interfacing functions falling under the responsibility of the A&E have a negative influence on acceptable operation of either the RSS or SHRS, then the PO will provide the direction to the LRM and A&E as necessary to reduce or eliminate this influence. Essential elements of this interface definition are illustrated in Figure C.1-1. The balance of this section is devoted to describing the interface features summarized in Figure C.1-1 and, where appropriate, providing the rationale backing the selection of the features.

An early action required of an RM involved in the implementation of the Reliability Program is a comprehensive review of the RM's internal engineering procedures. The purpose of this review is to identify all existing RM engineering procedures which must be updated to assure implementation of the

Reliability Program requirements. Modification of the engineering procedures makes compliance with the Reliability Program requirements mandatory and assures a Quality Assurance and program control overview of the program activities. Included in the listing of procedures which require modification are procedures relating to configuration control, design approval and supplier nonconformance/waiver approval. Modification of these procedures assures that Reliability organizations are involved in the approval of all aspects of the design, fabrication, shipment, installation, operation and maintenance of any item of equipment which is a part of or interfaces with the RSS or SHRS.

The procedures modifications outlined above are prepared by the RM Reliability Engineering group working with the RM Design Engineering group. In addition to the modification of existing procedures, implementation of the reliability requirements necessitates that a new procedure be generated. The new procedure requires and controls the generation and maintenance of a listing of the equipment and systems to which the Reliability Program requirements are to be applied. This listing is known as the Reliability Related Components List (RRCL). The list contains all items of equipment directly involved in the operation of the RSS and SHRS. Important interfacing, supporting systems are identified separately for appropriate reliability review.

Once the RM engineering procedure changes are in place, the Reliability Program requirements are placed on the RM Design Engineering organizations. The RM Design Engineering, Construction and Operation Organizations must then implement the Reliability Program requirements through all stages of design, procurement, fabrication, shipment, installation, operation and maintenance of the equipment. Verification of the correct implementation of the program requirements is obtained through the Quality Assurance review and audit activities. Additional verification is obtained by means of (a) a Reliability Engineering review of design review packages, waivers, nonconformance reports, etc. and (b) by the mandatory inclusion of a reliability review in the formal design reviews for each item of equipment on the Reliability Related Components List.

One of the special requirements imposed by the modified RM engineering procedures is for the production of a reliability document for each item of equipment on the RRCL. These documents are used as building blocks to construct a reliability evaluation of the total system.

Coordination of the Reliability Program output from the RM's is handled by the LRM. The LRM uses this output to assess the overall reliability of the RCS and SHRS. This assessment is used as an important indicator of the acceptability of the design of these systems.

All Reliability Program activities performed by the LRM and the RM's are subject to the direction and overview of the PO. The PO/LRM interface procedures used by the PO to direct and control the Reliability Program activities are similar in concept to those used to control the interface between the Reliability Engineering and Design Engineering groups within an RM organization. Details of the PO/LRM/RM interface are given in Figure C.1-2.

C.1.5 Appendix Content

The organization and content of the balance of this appendix are as outlined below. The intent of Sections C.2 and C.3 is to provide the background material necessary to place the subsequent Reliability Program description sections in the correct context.

Section C.2 provides a description of the analysis and testing techniques employed in the program.

Section C.3 contains a brief description of each of the systems included in the Reliability Program. The functions of the systems are described and the component parts identified. This section is provided as a convenience to eliminate the need for extensive reference to the main body of the PSAR.

Section C.4 provides the system designers' evaluation of their portions of the RSS and SHRS. Areas of system performance uncertainty are identified. A definition is provided of the Reliability Program activities necessary for the evaluation of the system performance uncertainties.

Sections C.5 and C.6 contain descriptions of the Reliability Program activities initiated to resolve the uncertainties identified in Section C.4. Results obtained to date from these activities are identified and discussed. The schedule for the production of results from activities still in progress is also discussed.

Section C.7 provides an assessment of the overall impact of the Reliability Program to date. Principal conclusions and design modifications stemming from the program activities are identified. The planned use of data from Reliability Program activities not yet completed is defined.

Addendum 1 contains a description of the test facilities for primary and secondary shutdown system tests.

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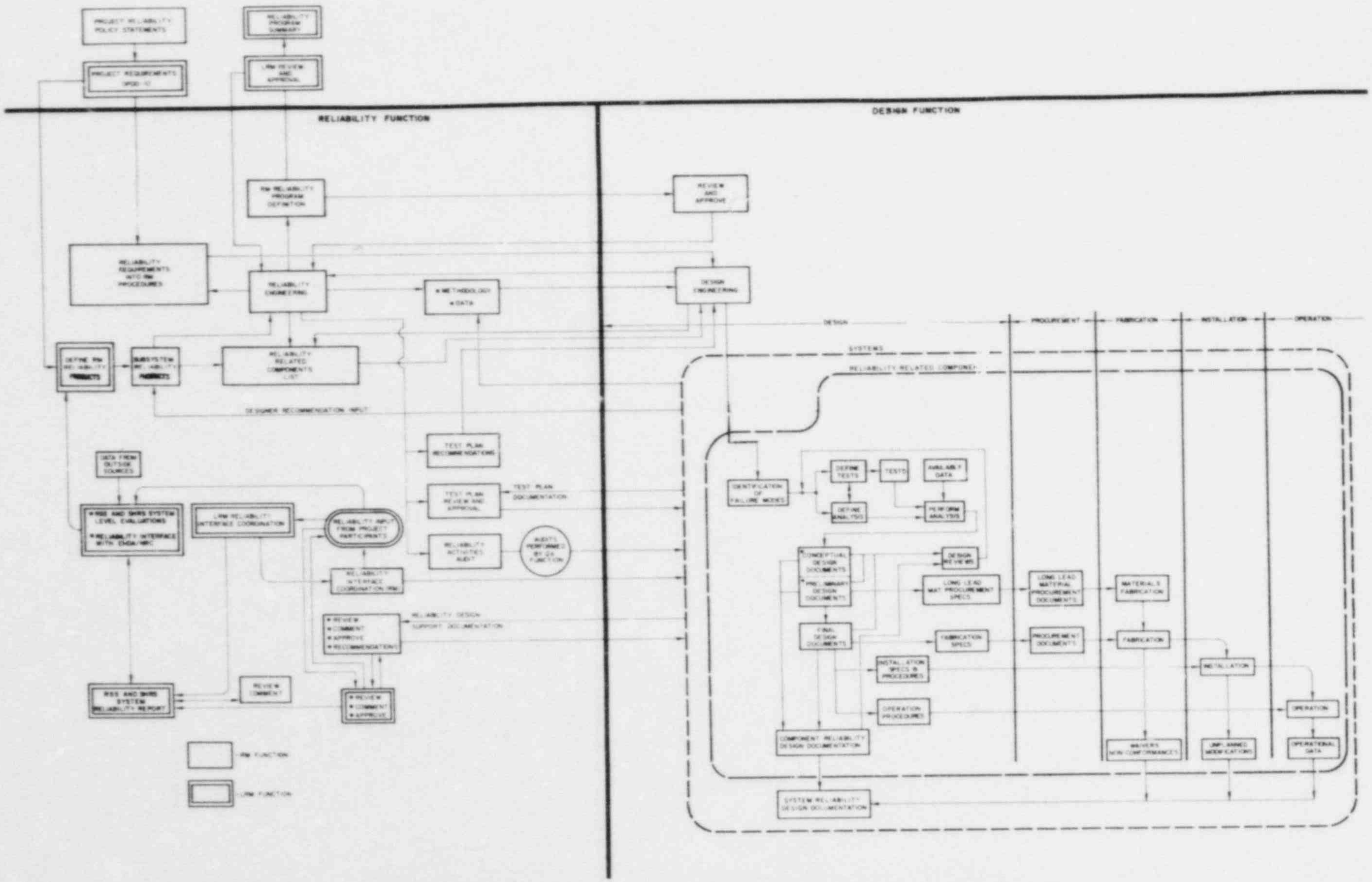

Figure C.1-1. RM Reliability Engineering/Design Engineering Interface Definition Chart

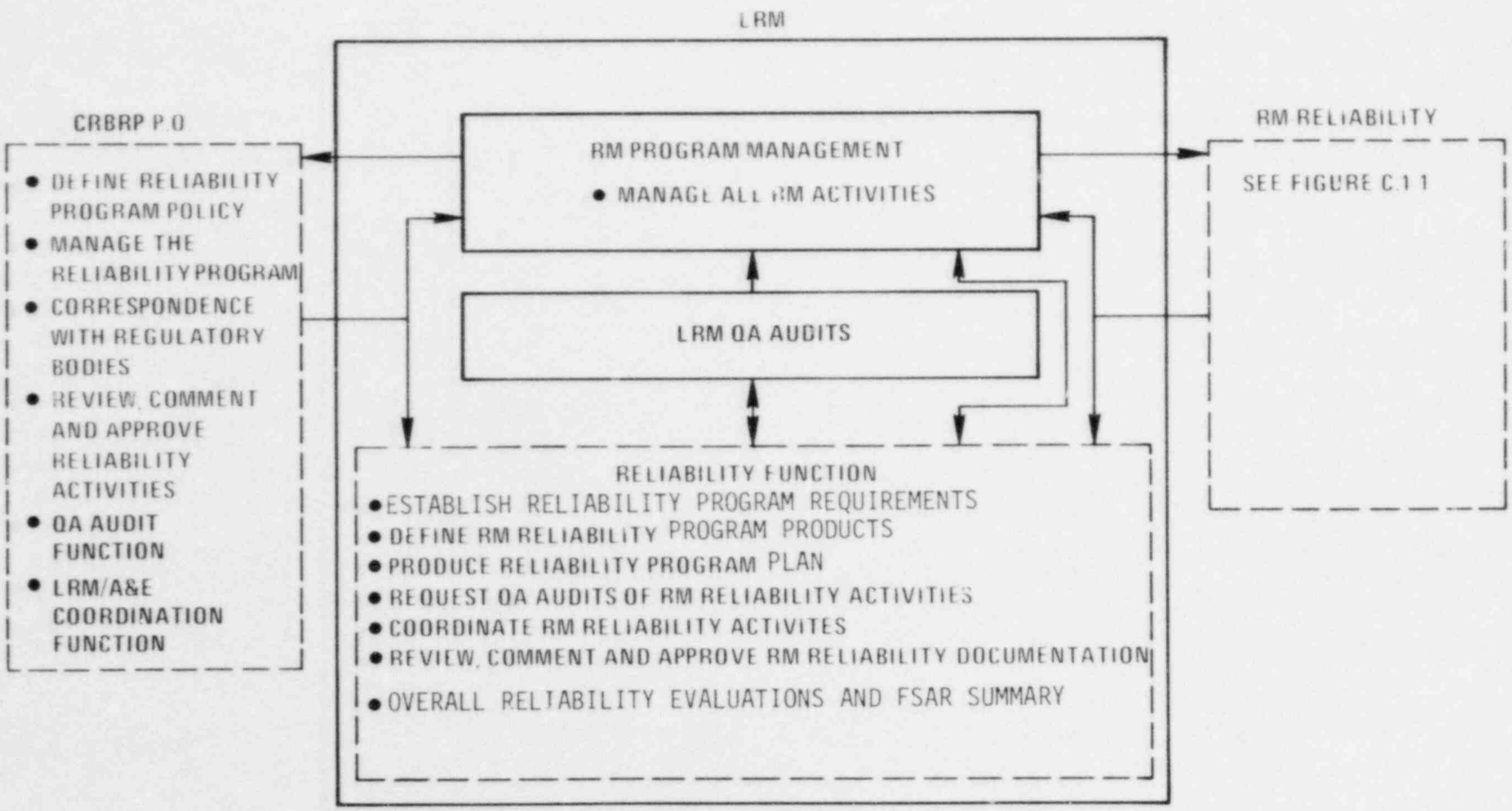

FIGURE C.1-2 PO LRM RM INTERFACE DEFINITION CHART

C.2.0 Program Guidelines

The purpose of this section is to review the process by which aspects of the RSS and SHRS having potential to degrade safety related reliability are identified and resolved. Conventional tools of reliability evaluation are being used to accomplish this task. These include Failure Modes and Effects Analysis (FMEA), common cause failure analyses (CCFA), testing and other methodology.

Figure C.2-1 shows diagramatically where reliability information is generated at each level of design and where this information is used. The remainder of this section provides a discussion of the activities shown in Figure C.2-1.

C.2.1 Component Level Evaluations

Failure Modes and Effects Analysis are the basic tool of reliability evaluation. They form the foundation upon which higher level evaluations are built. Failures critical to operational success are systematically identified and may be ranked according to both severity and probability of occurrence. These rankings are identified in Table C.2-1. Because of the technology (high temperature design, irradiation effects, etc.) involved in the design of individual components, it is essential that the component level FMEA be generated by the component design organization. To assure consistency of approach and continuity between component and system level evaluations, Reliability Engineering personnel are assigned to each component to support the evaluation. In several cases, a vendor with considerable experience in building the type of equipment under evaluation, (e.g., instrumentation and control devices, steam generator modules, etc.) was contacted to support generation of the FMEA. Typical output of the component level FMEA is:

- A comprehensive list of failure modes
- A list of potential causes
- Component designers view of the failure effect and criticality
- Initial estimate of the probability of occurrence

This output provides the initial assessment of design weaknesses and may result in immediate modifications (refer to Section C.7). It also is the first step associated with defining a test program. However, although estimates of criticality and probability of occurrence may be provided, they may be modified when aspects of redundancy and diversity are included in the system level evaluation. Because of the wide range of component types (structural members to electrical modules), the means for estimating probability of failure vary considerably. In some cases, meeting accepted code requirements may be deemed sufficient to indicate an acceptably low probability of failure. In other cases, considerable analysis may be required, especially if the failure has high criticality. Methods which are being employed to obtain failure probability estimates include stress/strength overlap, generic data and special testing designed to probe particular failure mechanisms. A further discussion of how estimates are made is provided in Section C.2.2.

All components which have the potential to impact successful operation of the RSS or SHRS are identified on a Reliability Related Components List (RRCL) produced by Reliability Engineering. An FMEA is performed for each of these components which then becomes the basis for categorizing the components according to failure mode effects. The reliability evaluation of each component is summarized in the Reliability Design Support Document at the component, component group, or system level.

Each component on the RRCL can be categorized as (a) degrading the functioning of the RSS or SHRS or (b) preventing the functioning of the RSS or SHRS. Those components on the RRCL are given a thorough review. Review of RRCL components includes all maintenance, shipping, installation and operation procedures, waivers and nonconformances as well as design documentation (specifications, drawings, design support documents, interface control data) and changes to those documents. Reliability Engineering is a participant in design reviews for all components on the RRCL.

The Reliability Design Support Document summarizes the activities performed to demonstrate achievement of reliability objectives. These reliability assessments are included as part of the equipment design support packages. The assessments are similar in character and stage of development to the other design support analyses in the package (e.g., stress, thermal/hydraulic, shielding, etc.). Reliability assessments include coverage and interpretation of all supporting development testing activities. Typical design support documents include (a) the FMEA's, (b) an assessment of critical failure modes to show design features to preclude or control the failure and (c) an assessment of common cause failure potential.

C.2.2 System Level Evaluations

System level evaluations are performed to relate detailed failure information to its impact on system performance. Just as a component designer is most qualified to assess the failure mechanisms associated with his component, the system designer must place each failure mechanism in perspective relative to overall system objectives. Although a component designer may indicate on an FMEA a high probability of failure or high criticality, when system considerations are included, these factors could change significantly. Some important considerations used to influence a change in criticality between the component and system level evaluations include:

- o Are other components which perform a similar function susceptible to the same failure mechanism?
- o Is the system in which the component is used redundant?
- o Can this failure initiate other failures that may affect the RSS or SHRS?
- o Can the component failure initiate failure of the entire RSS or SHRS?

A system viewpoint is especially important when determining if a design change is required to achieve adequate reliability. Both quantitative and qualitative evaluations are performed at the overall system level.

C.2.2.1 Numerical System/Subsystem Evaluations

Initially, a numerical evaluation is made based on random independent failure potential using component failure rate estimates. While it is recognized that random independent failure rates constitute only a part of the total failure probability, these assessments serve three important functions:

1) They provide an indication of the inherent system reliability if common cause failure potential is eliminated or controlled.

2) Components having greatest impact on predicted reliability are highlighted for priority attention in future evaluations.

3) They aid the designer as a decision-making tool for evaluating design changes.

The analytical techniques which have been applied to the system level numerical evaluations are success state, failure state and Markov modeling.

Success state or failure state modeling are techniques used to analyze a system's reliability on the basis of the system's and component's operational states. A logic block diagram is produced to display the system components and the logic associated with their configuration. From this diagram, different component failure combinations are evaluated to determine their effect on system operation. By evaluating all possible combinations of operational states of the system components that lead to successful or failed system operation, it is possible to derive an expression giving the reliability of the syste . This method has had application in both the RSS and SHRS. Details are reported in References 2 and 3.

To evaluate repairs or system reconfigurations during the time of interest (e.g., a full reactor operating cycle), Markov modeling has been selectively employed. All states of the system are defined along with the transitions that can occur between states. This information is mathematically represented by a series of first order linear differential equations which define the various states and the transitions into and out of each state. This method has been used in conjunction with success state modeling in the evaluation of the RSS. It is also being evaluated for use in the assessment of the SHRS.

Component failure rate data required to support the systems evaluations is obtained from many sources. Generic data used may be divided into two broad categories, directly and indirectly applicable to the components considered. Because certain elements of the RSS and SHRS are not unique to CRBRP, data exist which can be applied with little modification. This includes portions of the SHRS which are on the water side, as well as sodium side equipment which has been used in FFTF test facilities and sodium reactors such as SEFOR and EBR-II. In the RSS, considerable data exist on roller nut type control rod drive mechanisms which are very similar to the primary system mechanisms. Further, the design of the PPS electrical equipment is very similar to that developed and tested for FFTF. These directly applicable data are being used to the fullest extent possible and are discussed further in Sections C.5 and C.6. For other components, because their designs are relatively new, no significant failure rate data have been accumulated. It is therefore often necessary to derive component failure rate estimates from lower level piece part data. Military, industrial and governmental data sources are used in these evaluations (e.g., WASH-1400, MIL-HDBK-217B, etc.). These data may be modified using accepted reliability derating factor rules or engineering judgment. When modifications to data are made, the basis is made visible by thorough documentation.

Data available from components, subsystems and systems which are like the RSS and SHRS elements are being used extensively in the early assessments of the RSS and SHRS. This approach is particularly appropriate for the electrical systems because of the large bank of data available and the established acceptability of MIL-HDBK-217B methodology.

As the results from CRBRP test programs become available, these data are used to supplement those currently available. For reasons described later, much of the CRBRP testing will not yield failure rate data directly. Information obtained from some tests will be used to calibrate analytical techniques to provide greater assurance that reliability characteristics are correctly modeled.

C.2.2.2 Qualitative System/Subsystem Evaluations

Qualitative system assessments are used to determine system adequacy. Common Cause Failures (CCF) receive special attention because of their potential to significantly degrade RSS and SHRS capability. The first step toward elimination or controlling a specific failure mechanism is identification. Once the failure mode and potential causes are identified, corrective action can be defined. Resolution may take the form of elimination by design alterations, procedural control or demonstration of an acceptably low probability of occurrence.

Component level FMEA's are the starting point for CCF assessment. They provide a thorough listing of the failure mechanisms and associated causes for the elements of the system and its interfaces. FMEA's are a source of data for determining which system elements are susceptible to failure from common causes.

Past reactor operating experience plays an important part in the failure mode identification effort. Available operating experience is thoroughly reviewed to uncover common cause initiators underlying previous reactor incidents. The bulk of experience for hardware systems designed and fabricated to meet nuclear standards and codes lies in the domain of Light Water Reactors (LWR's). This body of data serves to identify potential component failure modes, design errors, operating problems and the actions necessary to correct these deficiencies. Output of the LWR incident survey includes:

- Review of reported incidents and identification of the information source

- Selection of incidents having CCF potential and identification of the causative mechanism

From these considerations, a list of potential causative factors is prepared which relate to LMFBR operating conditions. These causative factors are used as the basis for an evaluation of CRBRP component and system designs. This evaluation employs logic based on specific design features. This approach provides a systematic method for reviewing each failure and the relevant system functions required to successfully mitigate the incident.

As each failure is reviewed, protection provided against such factors is identified. The visibility of potential problem areas provided by the approach being employed assures that each design area is thoroughly investigated against the potential factors for common cause failures.

C.2.3 Testing

The CRBRP test program has as its objectives:

1) To identify to the appropriate design group potential failure points in the design of the components/systems

2) To assure the system design margins are adequate to meet the design specifications under the anticipated range of operating conditions

3) To determine the design margins against identified system weaknesses with the potential to degrade RSS or SHRS performance

4) To identify unknown system weaknesses

System level reliability evaluations were used to determine required testing. Factors evaluated to determine the need for testing include:

- Severity of the failure effect
- Common cause failure potential
- Estimated probability of occurrence
- Availability of applicable data from other sources
- Availability of verified analysis techniques for system performance evaluation

A primary purpose of testing is to demonstrate the capability of critical components to perform their function over as wide a range of operating conditions as practical. In some cases, tests will be run to failure to establish margins above those defined by the operational envelope.

Design performance and most postulated failures are affected by variations in the system operating environment. The fractional factorial design of experiments approach has been used in a number of instances in planning the test program to investigate the effects of variations in system environment.

Testing is performed at the component, subsystem and system level to explore failure modes of concern. Higher level testing provides maximum feedback of information concerning multiple failure modes and interface problems. RSS testing includes individual tests of the complete Primary and Secondary Control Rod Systems (drive mechanisms, driveline, absorber assembly and interface simulation). The electrical subsystem tests include essential system elements in a prototypic configuration. In the SHRS, testing above the level of individual components (e.g., steam generators, pumps, etc.) is accomplished at the plant start-up stage. Since the connecting elements (piping, wiring, etc.) are passive, component testing can provide a large portion of the information necessary to deterministically confirm system reliability. Supplemental testing and analysis is directed toward identifying and resolving potential interface induced failure modes. An example of such an activity is the piping integrity report, Reference 4. Where components, subsystem and system level testing identifies interfacing functions falling under the responsibility of the A&E which would degrade RSS or SHRS operation, then the PO will define any additional testing necessary to provide the needed level of information for use in resolving the identified degradation.

Accelerated life testing is employed to provide early feedback concerning potential failures. In the mechanical systems, this includes cyclic induced failures associated with the scram function under specified misalignments. Thermal cycling tests are performed on electrical subsystem equipment to accelerate the failure process involved in parts with latent time dependent failure mechanisms. Burn-in tests are used to screen out defective parts. Some failure mechanisms, however, cannot be investigated by performing accelerated life tests. For example, failure mechanisms related to time or operating conditions such as creep, self-welding and irradiation cannot be simulated by such tests. These failure mechanisms require separate tests, analysis or a combination of test and analysis for resolution. Time dependent failure mechanisms are evaluated whenever possible by operation of test hardware under prototypic conditions for extended periods of time. This section of the test program includes testing of complete Primary and Secondary Control Rod Systems under simulated prototypic operating conditions. In the SHRS, extended real-time testing of critical components such as the steam generator tubes will be performed.

Certain failure mechanisms will be explored by testing hardware that is not prototypic. However, this testing provides valid phenomenological information associated with these failure modes. Data obtained will be used to verify calculational models which are used to predict component behavior. Hardware and test fixtures specifically designed to explore the mechanism of concern are being constructed. Included in this category are irradiation effects, seismic induced loads and associated component interface characteristics, friction couples which influence unlatching and insertion, thermal striping and weld quality evaluation.

The impact of maintenance and operation on system reliability has also been considered in the design of the test program. Proposed CRBRP plant maintenance and operation procedures are employed whenever feasible in the tests. This includes equipment replacement activities that can be performed on both RSS and SHRS test hardware. Calibration procedures and repair actions associated with the electrical subsystem equipment are fully explored. Emphasis is placed on design feedback to reduce the potential of failure due to human factors.

Details of specific tests, rationale for those tests and expected outputs are described in Sections C.5 and C.6.

TABLE C.2-1

FAILURE RANKING CRITERIA

CRITICALITY RATING

Numbers*	Definitions
5	Failure to Perform Safety Function
4	Degradation of Safety Function
3	No Effect on Safety but Causes Unscheduled Outage
2	No Effect on Safety, Repair Deferred until Scheduled Outage
1	No Effect on Safety or Operation

PROBABILITY RATING**

3	An Off-Normal Condition Which Individually may be Expected to Occur Once or More During the Plant Lifetime
2	An Off-Normal Condition Which Individually is not Expected to Occur During the Plant Lifetime; However, When Integrated Over all Plant Components and Systems, Events in this Category may be Expected to Occur a Number of Times.
1	An Off-Normal Condition of Such Extremely Low Probability that no Event in This Category is Expected to Occur During the Plant Lifetime But Which, Nevertheless, Represent Extreme or Limiting Cases of Failures Which are Identified as Conceivable.

*Initial rankings are provided by the component designer and modified as appropriate in the system assessment.

**Alternative numbering schemes have been used on certain FMEA forms. The definitions of the categories are identical. In addition, an alternate approach which has been used is the actual estimated failure rates, obtained from the data base, manufacturer's specifications, pertinent literature, previous experience or tests.

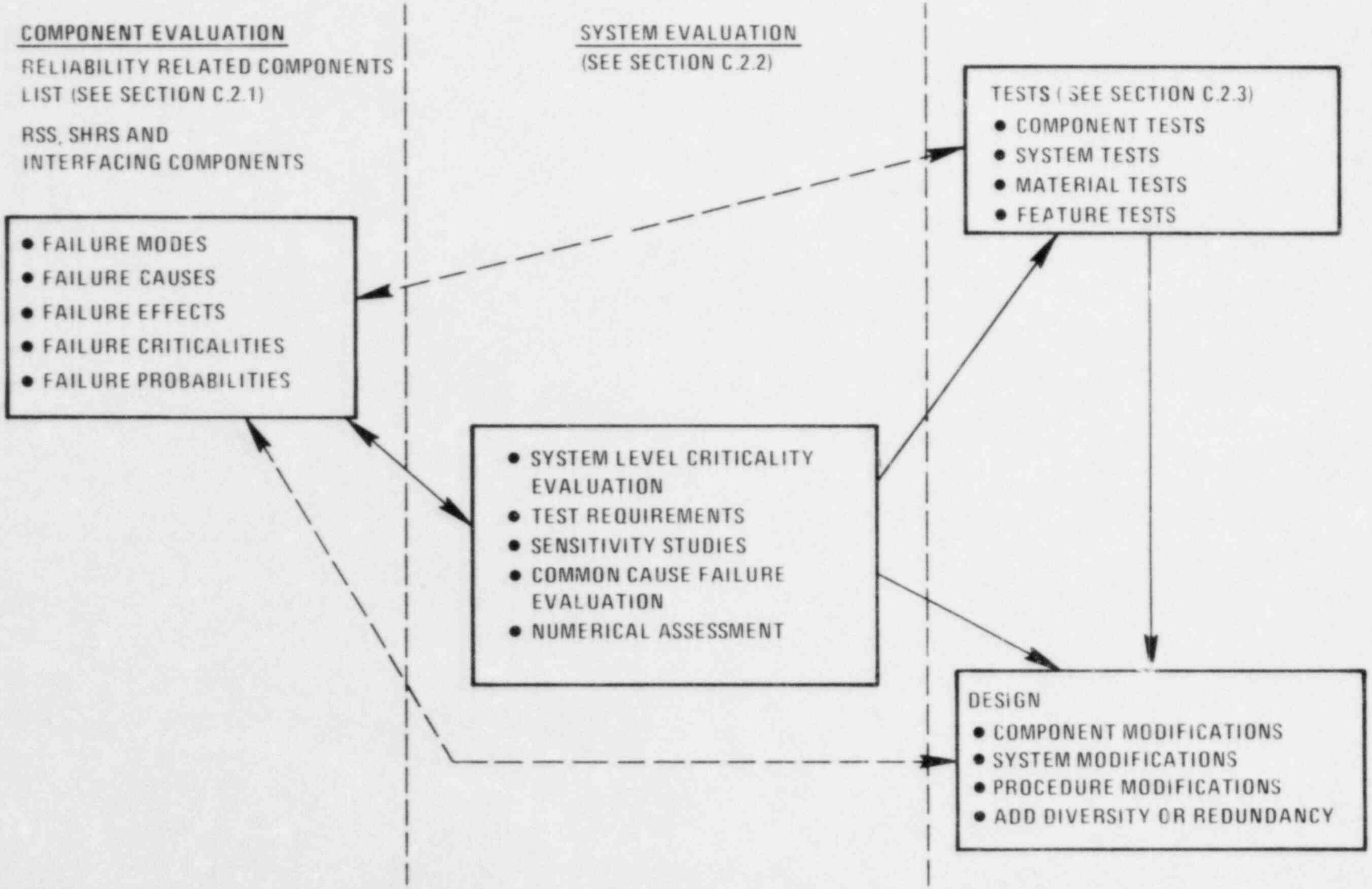

Figure C.2-1. Reliability Evaluation Activities

C.3.0 Systems Descriptions

This section provides a brief description of the functions and component parts of the systems included in the Reliability Program. This section is provided as a convenience to eliminate the need for extensive reference to the main body of the PSAR.

C.3.1 Reactor Shutdown System

C.3.1.1 Overall System Function

The Reactor Shutdown System (RSS) consists of two independent and diverse systems which are capable of shutting down the reactor without exceeding specified limits. (See Section 4.2 of the PSAR).

C.3.1.2 Design Description

The systems and components which make up the RSS are shown in Figures C.3.1-1 and C.3.1-2. A brief description of the systems and components follows:

Primary Mechanical Subsystem

The Primary Mechanical Subsystem (PMS) of the RSS includes 9 Primary Control Rod Systems (PCRS). Each PCRS consists of a Primary Control Rod Drive Mechanism (PCRDM), a Primary Control Rod Driveline (PCRD), and a Primary Control Assembly (PCA). The PCRDM is mounted on top of the reactor vessel closure head and provides mechanical actuation for insertion, withdrawal and scram functions of the control rod absorber. The PCRD connects the PCRDM with the control rod absorber. The PCRD passes through the upper internals structure. The PCA is located in the array of core assemblies and consists of a movable control rod (absorber pin bundle) and an outer duct assembly.

The PMS provides the functions of reactor startup, operational control and shutdown reactivity control. The primary function performed by the PMS which is reliability related is reactor shutdown (scram) for all conditions. Scram action is accomplished via disengagement of the roller nuts followed by downward motion of the control rod and driveline. Downward acceleration is achieved by means of the combined action of gravity and preload from the scram assist spring. All PMS functions are initiated by the primary electrical portion of RSS.

A. Primary Control Rod Drive Mechanism (PCRDM)

The PCRDMs are divided into two major sections which are described below:

The upper PCRDM assembly is an electro-mechanical actuating device which consists of a stator mounted on the outside of the motor tube and a collapsible rotor and roller nut assembly mounted inside the motor tube. The rotor assembly consists of a bearing mounted rotor tube and two pivoted segment arms. On each segment arm there are two roller nuts. When the stator is energized, the upper ends of the arms are pulled outward by the magnetic field and the lower arms are pivoted inward engaging the roller nuts with the threads of the leadscrew.

To produce a scram, the electrical power is removed from the stator causing the magnetic force field to collapse which releases the rotor segment arms. Springs separate the lower end of the arms and disengage the roller nuts from the leadscrew allowing the control rod to drop into the reactor core. A scram assist spring in the lower PCRDM is provided to supplement the gravity drop. A synchronizer bearing is provided to assure that both segment arms separate simultaneously. Anti-ejection pawls in the segment arms engage the leadscrew to prevent control rod ejection in the unlatched condition. These pawls are spring loaded allowing them to move out of engagement during downward motion of the leadscrew.

The lower PCRDM assembly consists of an extension nozzle, torque taker and tube, shield plugs, internal seal system and scram assist spring.

The extension nozzle is part of the pressure boundary and mounts the PCRDM to the intermediate rotating plug. The torque taker and torque tube constitute a torque restraint located in the space outside of the large bellows which prevents the mechanism leadscrew, bellows and PCRD from rotating. Keys on the torque taker slide in keyways in the torque tube over the full length of the stroke. The internal seal system utilizes three metallic bellows as well as conoseals to separate the rotor assembly and leadscrew from the reactor cover gas environment, precluding possible buildup of sodium frost on these components. The shield plugs provide radiation shielding for the PCRDM's and head access area.

B. Primary Control Rod Driveline (PCRD)

The PCRD consists of three concentric shafts: the driveline shaft, the disconnect actuating shaft and the position indicator rod. The driveline is the load carrying member and the outermost shaft. It connects the PCRDM leadscrew with the control rod. The disconnect actuating shaft (middle shaft) is used to disconnect the driveline from the control rod for maintenance or refueling. The innermost part is the position indicator rod which is used to verify that the control rod remains fully inserted during refueling, uncoupling and withdrawal of the driveline.

A dashpot is included in the PCRD to decelerate the driveline and control rod during the last few inches of insertion. The dashpot

The latch is located approximately five feet above the top of the active core region during power operation. The latch is used to grapple the coupling head and lift the control rod out of the core and to release the control rod for scram. The latch and the tension rod are the principal components in the SCRS for performing the scram release function. A short downward stroke of tension rod permits the latch grippers to move radially outward and release the control rod coupling head.

Two pairs of bellows isolate the SCRD internals and the SCRDM from the sodium vapor-argon cover gas environment below the reactor head.

C. Secondary Control Assembly (SCA)

The SCA consists of a movable control rod enclosed within a circular guide tube. The guide tube fits inside a hexagonal duct which is essentially identical to the fuel assembly ducts.

Internal flow paths are used to direct high pressure sodium flow against the hydraulic assist piston to generate the hydraulic scram assist force. Sodium flow used for scram assist is directed downwards through the circular absorber guide tube. Flow channels located in the control assembly nosepiece vent the scram assist sodium flow to the low pressure passages in the core support structure. Flow is also provided through the control rod pin bundle for cooling purposes. The bottom of the channel contains a nose piece which engages the high pressure plenum of the core support structure.

The control rod consists of a bundle of pins containing boron carbide (B_4C) as the neutron absorber. It is held in position above the core by means of the coupling head which fits into the latch of the SCRD. The control rod is free (when unlatched) inside the guide tube so that it will insert into the core by virtue of its own weight. Extra downward force is provided by hydraulic assist.

A damper mechanism is used to decelerate the descending control rod. The initial damping is provided by hydraulic dashpot action and the final portion of the descent is controlled by a hydraulic spring damper device.

For further descriptive and functional details, refer to Section 4.2.3 of the PSAR.

Electrical Subsystem

The Electrical Subsystem (ES) is part of the overall Plant Protection System. It consists of two independent and operationally diverse systems, the Primary Electrical Subsystem (PES) and Secondary Electrical Subsystem (SES) that monitor the condition of the plant and initiate scram of the primary and secondary control rods, respectively. Each system can independently initiate shutdown of the reactor. Each system has three redundant instrument channels and logic trains that provide sufficient redundancy to preclude degradation of either the PES or SES through a single failure.

The three redundant channels of each system are physically and electrically separated to assure that their independance is maintained. The ES is illustrated in Figure C.3.1-2.

Electrical piece/part requirements have been defined to assure high reliability. Most resistors and capacitors in the ES comparators, calculation units, logic and buffers are MIL-SPEC or established reliability components. Most transistors and diodes used in the comparators, calculation units, logic and buffers are qualified to Military Standard-S-19500. Most integrated circuits used are screened, inspected and tested according to Military Standard 883A, Method 5004, Level B. The vendor is also required to use Military Standard 454 as a specification for electronic module construction.

To provide added assurance against potential degradation of protection due to single failures, functional and equipment diversity have been designed into the ES. The PES responds to a different plant parameter than does the SES to provide protection against common cause failure of the sensing system. The only exception to the use of functional diversity is in subsystems which measure nuclear power. Since nuclear flux is the only parameter indicative of nuclear power that is fast enough to provide adequate protection, equipment diversity rather than functional diversity is provided in the power measurement system. Nuclear flux measurement is made in the PES using three compensated ion chambers and in the SES using three fission chambers. Section 7.2 of the PSAR includes lists of the ES protective functions and the design basis fault events and the first protective primary and secondary subsystem to respond to each event. The PES and SES instrumentation used to determine off normal conditions are also described in Section 7.2 of the PSAR.

The output signal from each of the three redundant sensors in each system is amplified and converted to a standard input signal by signal conditioning equipment. Where necessary, calculational units derive secondary variables from the sensed parameters. Where a single parameter is used in a level trip type, no calculational unit is used. Where ratio type trips are used, calculational units are used to derive the appropriate ratio. A comparator in each instrument channel outputs a trip signal when it senses that the instrument channel analog signal exceeds specified limits.

For additional diversity, the PES is configured using local coincidence logic while the SES is configured using general coincidence logic. In the case of the PES, each instrument channel outputs three redundant signals corresponding to either the reset (not trip) or trip state. Light emitting diodes and photo transistors are used to provide complete electrical isolation between redundant instrument channels and logic trains. The three redundant instrument channels are recombined as inputs to three redundant logic trains arranged in two out of three local coincidence. The 2/3 logic modules determine if two or more trip inputs are received from the subsystem comparators and then provide a trip signal to a 1/24 logic module. The 1/24 logic module outputs a trip signal if any of the 24 subsystems in a logic train have tripped. These signals deenergize primary scram circuit breaker undervoltage trip coils. The five primary scram circuit breakers are arranged so that when two or more logic trains trip, the scram circuit breakers remove power to the PCRDM's releasing the primary control rods. Manual shutdown and test capability is provided.

In the SES, each instrument channel comparator outputs a signal to the 1/16 logic corresponding to either the trip or reset state. The secondary logic system consists of the 16 protective subsystems arranged in a general 1/16 coincidence configuration. If any of the 16 channel A comparators trip, the 1/16 logic module inputs a channel A trip to the 2/3 con-
51 | figuration of each of the six SCRDM solenoid operated valves. Similarly, a trip signal from channel B or channel C comparators is transmitted to the SCRDM solenoid operated valves by the 1/16 logic module. The SCRDM solenoid operated valves are arranged in a 2/3 configuration such that a trip signal from two or more logic trains vents the latch cylinder, unlatches the control rod and allows it to be forced to its shutdown position.

C.3.2 Shutdown Heat Removal System

C.3.2.1 Overall System Function

Sensible heat in the structures and sodium and core decay heat are removed from the reactor following reactor shutdown by the Shutdown Heat Removal System (SHRS). The SHRS utilizes the normal heat sinks or alternate heat sinks to dissipate sensible and decay heat and prevent loss of coolable core geometry.

Normal heat removal paths are provided through three independent loops of the Primary Heat Transport System (PHTS) which transfer heat from the reactor to three independent loops of the Intermediate Heat Transport System (IHTS). Heat is removed from the IHTS by three independent loops of the Steam Generator System (SGS) to the main condensers. Alternate redundant heat sinks are provided through the Steam Generator Auxiliary Heat Removal System (SGAHRS) and the Direct Heat Removal Service (DHRS). SGAHRS provides an alternate heat sink for the main condensers and DHRS provides an alternate heat removal path and heat sink connected directly to the PHTS.

C.3.2.2 Design Description

The systems and components which make up the SHRS are shown in Figures C.3.2-1 and C.3.2-2. A brief description of the SHRS systems and components follows:

Primary Heat Transport System

The Primary Heat Transport System (PHTS) transports heat from the reactor to the Intermediate Heat Exchangers (IHX). The three PHTS loops transport the sodium coolant from the reactor vessel to the IHX's which connect the primary and intermediate loops. The three primary loops have common flow paths through the reactor vessel, but are otherwise mechanically independent and isolated in separate PHTS cells.

Each PHTS loop contains a hot leg centrifugal sodium pump, a permanent magnet flowmeter, a cold leg check valve and an IHX. Detailed descriptions of these components are contained in Sections 5.2 and 5.3 of the PSAR.

Intermediate Heat Transport System

The Intermediate Heat Transport System (IHTS) transports heat from the PHTS to the Steam Generator System. The system consists of three essentially identical, independent cooling loops operating in parallel to circulate sodium from the tube side of the IHX through the steam generators and back to the IHX.

Each of the cooling loops contains a cold leg pump, an intermediate sodium expansion tank, a permanent magnet flowmeter and piping to transport the sodium from the IHX outlet through the superheater and the two evaporators back to the IHX inlet. A detailed description of IHTS components is contained in Section 5.4 of the PSAR.

Steam Generator System

The Steam Generator System (SGS) extracts heat from the IHTS sodium. There are three independent SGS loops. Each loop consists of three steam generator modules (two evaporators and one superheater), a steam drum, a recirculating water pump, a Sodium-Water Reaction Pressure Relief System, a Sodium Dump System, a Water Dump System and a Leak Detection System.

The main Condensate and Feedwater System supplies feedwater to the steam drums. Superheated steam produced by each of the three SGSs loops is supplied to the single turbine generator. Feedwater is returned to the three steam drums from the condenser hot well by two condensate pumps and two of three main feedwater pumps.

The Sodium-Water Reaction Pressure Relief System (SWRPRS) becomes operational only in the event of a steam tube leak large enough to cause a rapid pressure rise from a sodium-water reaction. The system provides protection from over-pressure on the sodium side of the evaporator modules, superheater modules, IHTS and IHX by the use of rupture discs on the piping adjacent to the modules. The Water Dump System accelerates blowdown of the evaporator modules through quick opening water dump valves at the inlet to each evaporator module and reduces the extent of the sodium-water reaction. The Sodium Dump System provides sodium dump capability for the IHTS and the sodium side of the evaporator and superheater modules.

The Steam Generator Leak Detection System monitors for hydrogen and oxygen in the sodium in order to identify small leaks in the steam generator modules.

Details of the SGS components and subsystems are provided in Section 5.5 of the PSAR.

C.3.2.4 Electric Power Considerations

The sources of electric (AC) power for the SHRS are the preferred and reserve (off-site) AC power supplies and the standby (on-site) AC power supply. The standby power supply consists of two independent diesel generators and the emergency batteries and converter. Power under normal operation is needed for the primary and intermediate sodium pumps and for the steam generator recirculation pump and the main feedwater and condensate pumps.

For SHRS operation, standby power is supplied to the components of the PHTS, IHTS, SGAHRS and DHRS to assure operation in the event of loss of the main power supply. Standby power is provided to the two motor driven auxiliary feedpumps, the PACCs blowers, the pony motors for both the PHTS and IHTS pumps, the primary sodium makeup pumps, the EVST NaK and sodium pumps and the EVST air blast heat exchanger blowers. In addition, battery power is provided to the safety-related SGAHRS motor operated valves. The standby power supply is sufficient to facilitate and maintain adequate shutdown heat removal.

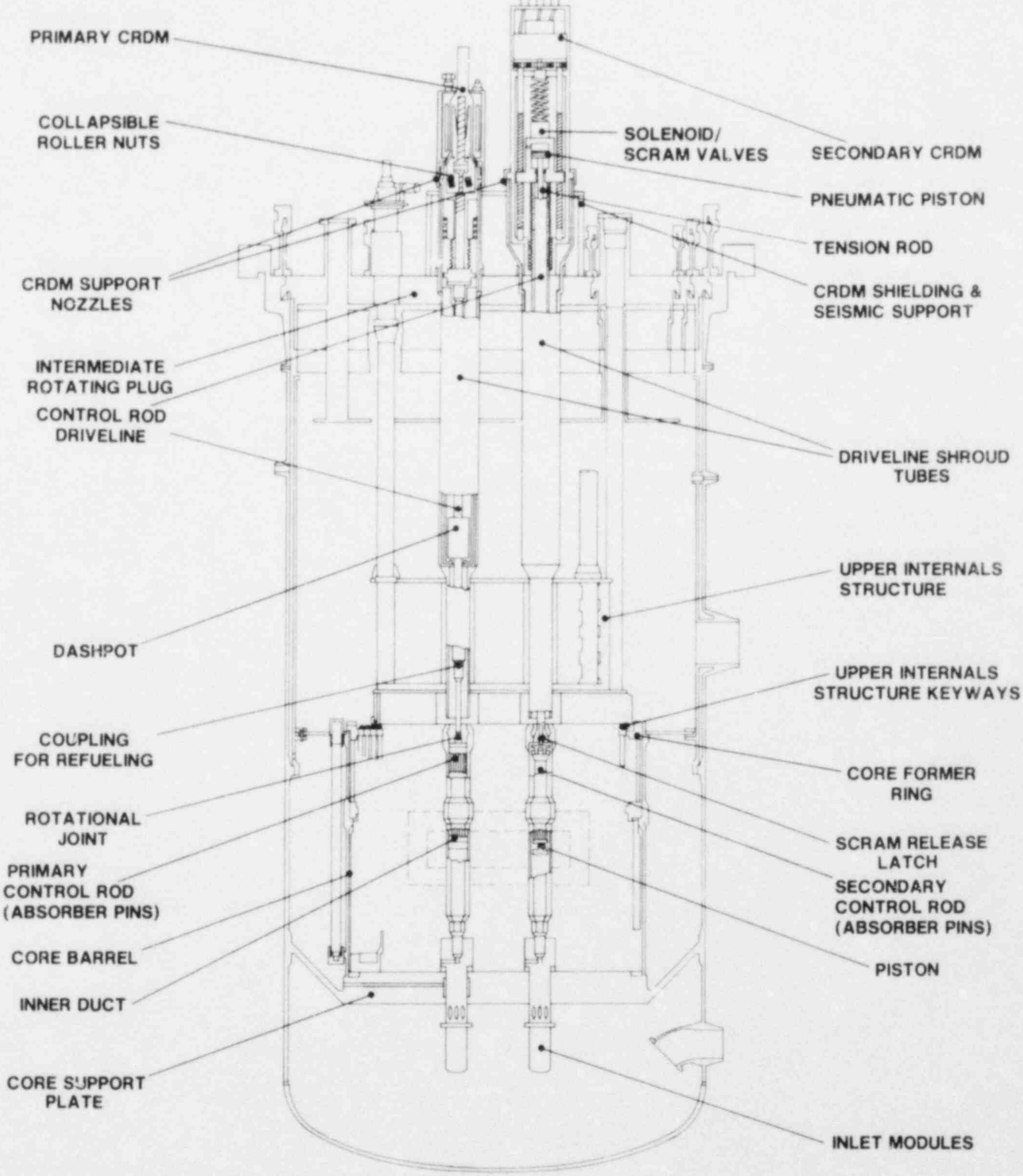

Figure C.3.1-1. Control Rod System Schematic-Reactor Elevation (Refer To Section 4.2 Of The PSAR For A Detailed Description Of The Control Rod Systems)

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C.4.0 Evaluation Focal Points

C.4.1 Reactor Shutdown System

This section provides the system designer's evaluation of areas of system performance uncertainty. Failure Mode and Effects Analysis (FMEA) coupled with FFTF testing and LWR operating experience were used to select the principal areas for reliability emphasis. The initial FMEA's were performed during the conceptual design stage. Identified potential scram failure modes and associated causes were correlated with available test data and analytical capabilities. Reliability emphasis was placed on credible failure modes having the greatest impact on scram reliability. Identified areas of uncertainty are cross referenced to specific tests which are described in Section C.5.

C.4.1.1 Primary Mechanical Subsystem (PMS)

This section describes the principal areas of the PMS identified as requiring reliability testing and analysis. Emphasis is placed on assessing areas which could cause failure of the PMS to perform its scram function. Common cause factors which could potentially lead to multiple PCRS failures and resultant shutdown failure of the PMS are identified for reliability emphasis. The PCRS scram function consists of the roller nut unlatching from the leadscrew and the translation of the leadscrew, driveline and control rod. Areas for reliability emphasis are identified separately for these two critical functions.

Areas Identified for Reliability Emphasis

A. Unlatching

The primary scram function of the PCRDM is to release (unlatch) the leadscrew upon loss of electrical power to the stator. The magnetic field of the stator holds the segment arms (rotor) radially outward which compresses the segment arm springs and engages the roller nuts with the leadscrew. With the removal of electrical power, the segment arm springs disengage the roller nuts and unlatch the leadscrew. PCRDM unlatching is dependent on segment arm spring forces and outward parting load forces generated at the leadscrew/roller nut interface. Forces retarding unlatching consist of the magnetic moment based on stator current decay, friction effects at the pivot pin and leadscrew due to roller nut contact, and inertia of the segment arms.

Areas for reliability emphasis to assure unlatching include investigations of: variations in friction coefficients, wear effects, manufacturing errors, internal misalignments, debris from leadscrew wear and relaxation or breakage of the segment arm springs. An analytical dynamic model is utilized to assure that adequate margins against these failure causes are included in the PCRDM design. Since FFTF testing has been highly successful and has provided a basis for analytical model verification, only prototypic PCRDM testing is considered necessary for reliability verification. PCRDM life tests beyond the design life are used to assess wear effects and to confirm operation. Tests at worst case temperature and pressure limits establish sensitivity of scram components to PCRDM environmental conditions. These tests are described in Sections C.5.1.2A and B.

Since the PCRS is required to scram during seismic events, verification of unlatching capability during seismic excitation is necessary. Relative lateral motions of the leadscrew and roller nuts, vertical acceleration and potential structural failures are the key areas for reliability evaluations. Shaker tests are planned to assess seismic performance and support analytically determined margins for seismic conditions. The PCRS seismic test is described in Section C.5.1.2I.

If PCRDM bellows leakage occurs, sodium vapor can enter the upper mechanism. Sodium vapor can affect the unlatching function by increasing friction coefficients and sodium solidification can occur in tight clearances between moving parts. A failed-bellows test will be conducted to confirm acceptability of operation with a failed bellows. The failed-bellows test is described in Section C.5.1.2B.

The PCRDM/PCRDs are given acceptance tests at the vendor prior to shipment. In addition, operational tests are performed prior to initial CRBRP startup and scram tests are performed after each refueling. These tests provide a basis for identification of manufacturing, installation and maintenance errors affecting scram performance.

Human errors could occur during maintenance operations and could adversely affect scram performance. To minimize the potential for and the effect of maintenance errors, maintenance procedures and tools will be developed in PCRS tests. Prototypic maintenance operations will be performed throughout the PCRS tests to identify any effects on system performance.

B. <u>Insertion</u>

The PCRS scram insertion function involves full stroke and partial stroke motion of the connected leadscrew, driveline and control rod. The areas where contact points and minimum clearances could affect scram insertion are: the leadscrew to PCRDM upper and lower bushings, the PCRD to lower PCRDM torque taker keyway, the PCRD shaft to the bottom of the dashpot cup, the PCRD piston to dashpot cup (over the last few inches of insertion), the control rod shaft coupling to the PCA scram arrest flange (over the last portion of insertion), the control rod wear pads to the PCA outer duct and the control rod inner duct to the PCA outer duct. In a seismic event, additional contact points may occur between the PCRD and the PCRDM shield plug and the PCRD and the driveline shroud tube. Areas for reliability focus are the effects of misalignments, seismic loadings, friction coefficient variation, wear, irradiation and manufacturing errors.

PCRS misalignments are established from manufacturing and installation tolerances and clearances between PCRS parts and interfacing reactor system components including the reactor vessel closure head, reactor vessel, core support structure, core barrel with associated core former rings of the core restraint system, core assemblies and the upper internals structure. An analytical model is utilized to evaluate PCRS performance under varying misalignments for confirmation of scram insertion performance. Analysis of the interaction loads resulting from control rod system misalignments is a complex process however, involving three dimensional mixed structural and mechanical response of the interfacing components and the driveline/control rod assembly. Existing structural analysis tools require the application of engineering judgments to deal with mixed mechanical-structural response of complex systems. To verify these engineering judgments, test calibration of the analytical models is required. Misalignment tests of PCRS performance are utilized to confirm scram capability.

Lateral seismic accelerations leads to a "rattling" effect on the driveline with impulsive drag forces resulting from impact of the driveline/control rod with the surrounding structures. The frequency and magnitude of the impact forces are influenced by fluid coupling between the driveline and guiding structure, squeeze film fluid effects at impact and dynamic friction coefficients. PCRS testing under conditions of simulated seismic excitation will establish the magnitude of these effects and permit calibration of the seismic scram insertion analysis. The PCRS dynamic seismic friction test is described in Section C.5.1.2C, and the PCRS seismic test is described in Section C.5.1.2.I.

Lateral loads on core assembly outer duct load pads during a seismic event could lead to control assembly duct deformation or increased PCRS misalignments. The above core load pads are located in a region where substantial irradiation induced ductility loss is anticipated. Brittle fracture of the load pads must therefore be considered as a potential failure mode. A duct crushing test in support of CRBRP core assemblies has been completed. Analyses of this test, which showed no brittle failure of irradiated ducts under loading conditions prototypic of CRBRP seismic loads, will be used to confirm PCA design margins. The duct crushing test is described in Section C.5.1.2L.

Wear effects between moving parts can lead to changes in scram speed as a result of changes in effective friction coefficients and clearances. Galling of sliding surfaces can lead to significant friction increases and to seizing of moving parts. Wear effects can best be evaluated by tests under prototypic conditions. PCRS testing in sodium loops is planned to evaluate wear effects on system performance. Testing exceeding the design basis service life (number of scrams, feet of travel) will be performed to establish lifetime margin relative to wear effects. Tests to investigate wear effects are described in Sections C.5.1.2A and B.

Control assembly outer duct galling was observed in FFTF testing as a result of forced contact (beyond design basis) between the movable pin bundle wear pads and the outer duct. The pattern of galling marks observed was consistent with a torsional loading transmitted from the CRDM to the control assembly absorber section. A rotational joint has been included in the PCA design to minimize PCRDM torque transmission and the resultant inner/outer duct contact. Testing to verify rotational joint performance in sodium has been satisfactorily completed. The rotational joint test is described in Section C.5.1.2F.

Irradiation effects on PCRS scram performance must be considered in the PCA. Irradiation could degrade scram performance by reducing control rod to outer duct clearances either as a direct effect of irradiation induced swelling or as indirect effects of duct bowing. Reduction in clearances resulting from failures of component parts due to ductility loss or inner duct distortion from pin bowing or pin ruptures must also be considered. Bowing of the inner and outer ducts results from thermal and flux gradients across the ducts leading to differential irradiation swelling combined with creep effects from interactions with adjacent assemblies. Section 4.2.3 of the PSAR provides a detailed discussion of duct bowing. Differential bowing between the inner and outer ducts could, if the bowing magnitude were sufficiently large, lead to duct to duct contact with resultant increase in drag forces retarding scram insertion. CRBRP programs to verify swelling and creep correlations used for bowing analyses are underway. These programs together with operational data from other reactors (FFTF, EBR-II, etc.) will provide a basis for establisning the magnitude of duct bowing. Analyses have been conducted to assure that design envelope duct bows do not result in significant drag forces and that margins exist against scram failure. A duct bowing test using prototypic ducts has established scram limiting duct bows and determine duct bowing margins.

Absorber pin ruptures could impact scram performance due to reactivity loss from B_4C washout at open cladding areas or due to inner duct deformation from pin failure gas pressure pulses. Absorber pins are designed not to fail and analyzed using conservative deterministic cladding criteria. Washout test data for B_4C pellets exposed to flowing sodium must be evaluated. This evaluation is described in Section C.5.1.1. Reliability emphasis is placed on pin rupture effects of mechanical deformations that result from sudden release of pin internal pressure. Tests have been performed to determine inner duct deformation (ballooning effect) from pin failure pressure pulses to envelope worst case effects of pin rupture. These tests have shown that pin ruptures do not result in significant deformations and have a negligible impact or scram performance. The pin rupture tests are described in Section C.5.1.2J.

Temperature and flux gradients across an absorber pin produce pin bowing. Due to dominance of the temperature gradient with higher temperatures at the interior side of the pin bundle, the bow is nominally inward. With wire wrap point load constraints, local outward deflections of the pins will occur. This outward deflection coupled with pin swelling can close pin to inner duct clearances and result in an outward pressure on the inner duct. The inner duct is necessarily a thin wall member. It could therefore deflect outwards under the action of pin bundle/duct interaction loads. Pin deflection effects on the inner duct could have an adverse impact on scram performance due to a reduction in duct to duct clearances. Analytical predictions of pin to pin and pin bundle to duct interactions are made. Data are available on this type of interaction behavior from FFTF fuel pin testing. While not prototypic, this provides additional support to the analysis for the current design. The behavior of the pin bundle is characterized by a mixture of structural and mechanical response to the applied loading. Models using existing structural analysis tools are however subject to uncertainties. A pin compaction test has been conducted to determine inter-pin loads, pin to duct loads and pin bundle compressibility to envelope the inner duct deformation that might result from pin bowing. Testing described in Section C.5.1.2G has provided the data required to define the pin bundle—duct interaction analysis.

The impact of the driveline/control rod on the PCA scram arrest flange must be evaluated to verify that a brittle fracture of the irradiated PCA outer duct will not occur. Transmitted and reflected (from core support structure) stress waves can lead to a stress buildup and brittle failure of the irradiated duct becomes a potential failure mode. The reason for concern with this failure mode is that potential chips or duct distortion from the duct fracture could retard scram insertion. To support analyses in this area, duct impact testing was conducted to minimize analytical uncertainties and to provide test confirmation that brittle fracture will not occur. The duct impact tests are described in Section C.5.1.2K.

The previously identified PCA component testing will provide data at an early date which is directed at specific areas of reliability emphasis for feedback into final PCA design. Based on preliminary analyses, testing is anticipated to confirm a design lifetime of two years. To obtain direct irradiation behavior data to confirm PCA lifetime capability, irradiation testing of a PCA in FFTF is planned. Post irradiation analysis will provide direct data on duct bowing, pin pressures, irradiation induced swelling, etc. prior to extended power operation in CRBRP.

Manufacturing, maintenance and procedural errors could affect scram insertion through factors such as internal misalignments of the assembled components, incorrectly assembled joints or materials errors. These factors could result in increased insertion drag forces resulting in slowed scram insertion. Vendor acceptance tests and development tests on prototype and plant manufacturing units will be performed. These tests will make maximum practical use of common materials for both test and plant units.

C. Interfacing Components

Interfacing component failures can impact PMS scram performance by increased misalignments, temperature effects and flow changes. To ident.ry failures, failure detection capability and effects on PMS performance, analysis of interfacing component failure modes and their effects are performed. This analysis together with PCRS performance evaluations against identified failure modes is directed at reliability enhancement.

C.4.1.2 Secondary Mechanical Subsystem (SMS)

The SMS has been evaluated from a reliability standpoint and the continuing reliability efforts will concentrate on the safety related functions, i.e., the unlatching and insertion of negative reactivity into the core (scram). This section describes those features of the SCRS where design and reliability efforts are emphasized.

Areas Identified For Reliability Emphasis

Each of the three components of the SCRS has a function which is necessary for successful scram performance. The SCRDM scram function is to release the pneumatic holding pressure thereby allowing the tension rod to fall. The SCRD function is to allow the tension rod to drop a fraction of an inch to open the gripper fingers. The functions required to assure successful scram are control valve and piston/cylinder operation, tension rod translation, latch release and control rod insertion. Each of these is discussed pertaining to factors involved in the SCRS design and environment and areas of expected reliability activity.

A. Control Valve, Piston/Cylinder Operation

A pneumatic actuator, connected by the tension rod to a gripper device latched to the control rod, is vented when electrical power is cut off to at least two of the three solenoid operated control valves. Venting of the actuator allows the tension rod to drop unlatching the gripper device from the control rod coupling head.

Reliability Program activities on the control valves center on eliminating any potential for jamming, sticking or slow operation. The possible long term hold periods (up to one year between operations) could result in valve degradation. Mechanical distortion caused by thermal effects or shock impact in the valve assembly could cause binding forces in the pilot valves, solenoids or main valves. Deposition of particulates in the pressurizing gas could also lead to binding as well as to port and/or vent blockage. Thermal degradation of valve seat material or the presence of gas contaminants could jam or delay the operation of the valves and armatures. Variations in friction coefficients, wear effects on the valves, manufacturing errors and internal misalignments could impact proper valve operation. The effects of galling and wear and the potential sodium vapor effects (caused by bellows failure) on clearances are areas of reliability emphasis.

Verification tests (see Section C.5.2.2J) will provide proof-of-principle for the hydra[illegible] assist and control rod insertion. Guide tube deformation tests will [illegible] used to determine the effects on rod insertion. These tests are described in Section C.5.2.2H. Guide tube distortion beyond the maximum calculated value will confirm that performance margin exists. SCRS life tests wil' be used to provide wear data under prototypic environmental conditions.

E. Interfacing Components

The successful scram function of the SMS could be affected by the reactor components which interface with the SCRS's. Interfacing components could potentially apply loads to the SCRS or allow displacements of SCRS components beyond design envelope misalignments which could prevent scram. Examples of components which are in these two categories are the upper internals structure, the reactor closure head, fuel assemblies and the core support structure. Analysis of each interfacing component to determine its failure modes and their effects (FMEA) is used to identify areas where adverse effects could exist. Analyses and test results will be used to provide substantiation that interfacing components are not potential scram failure initiators. Where the analyses indicate potential initiators, the component design will be modified to remove or minimize the potential.

C.4.1.3 Electrical Subsystem (ES)

The systems and equipment covered in this section comprise electronic and electrical signal conditioning equipment with associated cabling, instrumentation and switchgear needed to operate the mechanical shutdown subsystems.

System design features included in the CRBRP ES are similar to those widely used in LWR's. These equipments are implemented using piece/parts which have been proven in military programs. The equipment designs for many of the electronic subsystems, while based on designs for FFTF, have not been proven in an operating environment. As a consequence, reliability analyses and test programs are aimed at providing the same confidence in operational reliability as now exists for similar equipment in LWR's.

To evaluate the reliability of the operation of the ES, the ES is subdivided into three major areas of interest: overall subsystem, instrumentation sensors, and electronic components and subsystems. These areas are discussed in turn. The principal considerations affecting reliability are reviewed along with the resulting conclusions concerning any additional needs for analysis and testing.

A. <u>Overall Subsystem</u>

The Plant Protection System (PPS) design has a close similarity to systems used in Light Water Reactors for which there is an extensive background of standards, regulatory guides and licensing practice aimed at improving safety. The basis for the PPS design, the standards used and the supporting analysis are described in Chapter 7 of the PSAR and are at least as stringent as those applicable to PWRs. The use of these standards, coupled with the similarity of CRBRP and LWR designs for the ES, is a major factor contributing to the program goal of achieving a level of reliability in each of the ES subsystems for CRBRP comparable to that achieved in LWR systems.

Reliability is assured by a combination of design procedures, tests and system reviews which ensure that the requirements of the standards have been adequately met both within the ES subsystems themselves and with relation to other interfacing systems and equipment.

Assurance of reliability in design features within the ES has been met by a combination of studies of failure modes and effects (in order to determine the results of single failures) and of common causative factors which could result in total system failures.

The industry standards referenced in Table 7.1 of the PSAR provide for the reliable operation of the ES equipment under accident conditions by placing specific requirements for separation, environmental qualification and testing. For instance, in the case of separation of ES from others, full compliance with Regulatory Guide 1.75 is required in all relevant system design descriptions. This has been implemented in the case of the ES by specially designed buffer circuits, by requirements for appropriate cable and tray separation and finally by the use of separate upper and lower cable spreading rooms for the primary and secondary electrical subsystem cabling respectively.

The activities described previously are directed at assuring that reliability is designed into the ES. The effectiveness of these design measures is assessed in the ES reliability assessment. This assessment is conducted using the techniques described in Section C.2. Particular attention is directed to the reliability evaluation of many ES interfaces with external components. The ES reliability evaluation is summarized in the ES Reliability Design Support Document.

Given the use of existing proven industry standards and design and reporting procedures that ensures their effective implementation, difference remains between CRBRP protection systems and those currently licensed for LWRs. The difference relates to the design of the protection logic systems. Although these systems are based on the FFTF design, no overall system operating experience exists for this equipment. To provide the required operating experience for the CRBRP system, the Reliability Program plans long-term testing of a complete Electrical Subsystem. A description of this test is contained in Section C.5.3.2B.

B. Instrumentation Sensors

Instrumentation sensors to be used in the ES are based on designs in which there is extensive previous experience from either Light Water Reactors or other sodium systems such as EBR-II, Fermi-I and SEFOR. For instance, in the case of the Power/Flow trips on which reliability interest has been concentrated, there are four different types of sensors involved: Neutron Flux Ion Chambers, Neutron Flux Fission Counters, Electromagnetic Flow Meters, and Sodium Differential Pressure Meters. The two types of neutron flux sensors are similar in both construction and functional application to sensors used in Light Water Reactors. Their reliability characteristics are consequently well understood and can be factored into the overall system design by means of accepted and proven redundancy concepts.

The electromagnetic flowmeters are based on instruments in which operating experience exists in EBR-II and Fermi and operational sodium test loops. The design utilized electrodes connected to the outside of pipes and permanent magnets located again outside the pipes. The simplicity of these sensors provides inherent high reliability.

In the case of differential pressure sensors, the similarity of the instrument to those used for similar functions in operating sodium test loops provides assurance that their failure characteristics are well understood.

In view of these considerations, it was concluded that an adequate background of relevant experience existed on the sensors to support their reliable operation in the CRBRP ES application. This conclusion also applied to the sensors which initiate the shutdown heat removal system. These sensors are similar to those now in operation in similar applications in the industry.

C. Electrical Components and Subsystems

The CRBRP ES design was based on a modification of the design prepared for the FFTF project. A reliability enhancement study carried out early in the program indicated what improvements could be most effectively achieved by means of a component reliability program. This component reliability activity has taken the form of rigorous comprehensive specs which include extensive use of MIL Specs and a component test program.

The component test program has two principal series. First, vendor thermal screening and functional tests will be used to detect any design or manufacturing deficiencies in the modified FFTF components. Second, extended life tests will provide a high confidence level in the long-term reliability of the components.

C.4.2 Shutdown Heat Removal System

This section describes the areas of the Shutdown Heat Removal System (SHRS) where testing and analysis have been identified as desirable to support the adequacy of SHRS reliability. These areas relate to uncertainties associated with specific components and have been identified after an evaluation of the overall system. Included are separate sections on the PHTS, IHTS, SGS and SGAHRS, and DHRS.

Failure Mode and Effects Analysis, numerical reliability predictions for conceptual and preliminary design configurations and designer experience with both sodium and water/steam systems and components were used to select the initial areas for reliability attention. At the initial stage in the Reliability Program, major development test programs existed in the steam generator systems and coolant boundary areas of the heat removal systems. These areas were considered for reliability emphasis since defined tests will provide information which could impact the reliability of the SHRS. The initial FMEAs and reliability assessments were performed during the conceptual and preliminary design phase. The failure modes and failure consequences and simplified systems reliability models were used to determine relative criticality of failure modes. During the design detail phases, the FMEAs and CCFAs are upgraded to reflect design maturity and changes. These evaluations confirm the appropriateness of ongoing heat transport system component development tests. Identified areas of uncertainty are cross-referenced to specific tests which are described in Section C.6.

C.4.2.1 Primary Heat Transport System (PHTS)

Areas Identified for Reliability Emphasis

The CRBRP PHTS design basis and operational environment are similar to those of FFTF. The primary pump and IHX designs as well as piping layout are areas of major difference. FFTF experience in design, fabrication, shipping, installation, inspection and operation will be utilized in the CRBRP reliability evaluations and in final design implementation.

One area of criticality to PHTS function during shutdown heat removal identified for reliability emphasis is the structural integrity of the primary coolant boundary. The FFTF experience will be significant to assessing the reliability adequacy of this boundary and identifying appropriate activities for assuring its installed integrity. Retention of the primary system coolant inventory has critical importance to transporting heat from the core. Coolant boundary integrity is also important to the successful operation of DHRS since some piping or vessel leaks may lower sodium levels below the sodium overflow level which would terminate DHRS removal of heat from the reactor vessel. Loss of primary system coolant inventory is limited by guard vessels and elevated loop piping provided to maintain independence of primary loops. Leaks in one PHTS loop will not affect inventory in the other two PHTS loops. Test programs have been directed toward assuring adequacy of base material (SS304, SS316 and Inconel 718) structural properties and the welded joint design adequacy. Much of this information is being developed under on-going technology programs described in Sections C.6.1.2H and L. Present programs may not provide sufficient information on weld joint reliability and thermal fatigue at locations where sodium streams with widely differing temperatures mix. Testing in these areas is described in Section C.6.1.2A.

Pipe hangers and snubbers impose loadings on the piping and represent potential failure mode initiators for the primary sodium boundary. Analyses of seismic response and expansion characteristics assuming failed hangers and snubbers are being performed. Testing to qualify the hangers and snubbers will be performed. The testing is described in Section C.6.1.2B.

The PHTS sodium leak detection system is a fundamental line of defense in the assurance of the primary coolant boundary integrity. Development programs are in place as described in Section C.6.1.2D which will provide diverse sodium leak detection methods and equipment with appropriate levels of sensitivity.

The PHTS pumps provide forced circulation and are important contributors to the overall reliability of the SHRS. Depending on the time after scram that the DHRS may be activated, operation of one or more of the primary pumps at pony motor speed is critical to the operation of the DHRS. Low speed pump tests have been specified. The primary pump development program includes tests on pony motor operation and will provide information on pump bearing wear characteristics at pony motor speeds. These tests are described in Section C.6.1.2F.

The main heat transport system is designed to provide natural circulation heat removal capability in all three loops. Testing is planned to confirm operation in this mode and is described in Section C.6.1.2G.

C.4.2.2 Intermediate Heat Transport System (IHTS)

Areas Identified for Reliability Emphasis

The functional requirements for the IHTS and its hardware characteristics are similar to the PHTS. Therefore, PHTS materials properties testing, leak detection testing and sodium pump testing are applicable to IHTS reliability assurance activities. The impact of intermediate system sodium leaks introduces new variables for attention. The ambient air environment and the material property differences in the transition welds to the SGS and the extensive length of piping runs are key areas of difference between the IHTS and PHTS. The ambient air environment for the IHTS introduces an increased level of corrosion potential around a small sodium leak. Sodium leak detection testing must therefore be directed towards sodium to air leakage. The materials property testing for the PHTS, however, is applicable to IHTS reliability assessment.

The IHTS connections to the steam generator introduce two unique areas of structural design. The transition weld joint at the piping connections to steam generator modules and the mixing tee joints which tie the modules in one loop back to a single pipe are areas with potential for loss of coolant boundary integrity. Development tests are underway for these two areas to investigate their potential for being a point of coolant boundary failure. These tests are described in Section C.6.2.2.

C.4.2.3 Steam Generator System (SGS) and Steam Generator Auxiliary Heat Removal System (SGAHRS)

Areas Identified for Reliability Emphasis

The steam generator systems have common elements in both the normal shutdown heat removal mode, which uses the main steam piping and condenser as a heat sink, and the auxiliary heat removal mode which uses SGAHRS to provide steam venting and PACCs as heat sinks.

There are nine steam generator modules of common design, any one of which is adequate to remove shutdown heat. The designs potential for common cause failures in the modules and their associated systems, and steam generator coolant boundary integrity is a primary focus of reliability activities. Steam generator module tests are described in Section C.6.3.2A.

A shell-side hydraulic model test is providing information on the potential for tube or tube sheet vibration in addition to shell-side flow distribution. The "few tube" model tests provided information about tube expansion during thermal transients. The potential for thermally damaging the steam generator tubes as a result of departure from nucleate boiling (DNB) is also being experimentally investigated within the steam generator development programs by exposing tubes to severe DNB conditions. Descriptions of the steam generator prototype test and "few tube" test are provided in Section C.6.3.2A.

The leak detection system has the potential for improving the availability of steam generator modules for shutdown heat removal. This system is therefore of interest to reliability. The leak detection system signals that hydrogen or oxygen is present in the intermediate system sodium. The operator would take action to isolate the water/steam side and may take action to dump the sodium from the affected loop. Such action would remove the loop from heat removal capability. Early action by the operator may preserve the sodium inventory by isolating the water/steam side. The early action may retain the loop for heat removal through the unaffected modules. Testing has been defined in support of leak detection function and is described in Section C.6.3.2B.

The burst discs in the SGS which isolate the SWRPRS from the SGS are receiving major attention since they introduce a common cause failure potential for the three main heat transport systems. Inadvertent rupture of one pair of these discs in each loop would eliminate redundancy in the SHRS (only the DHRS would then be available). The SGS development program is conducting tests to demonstrate operation of the burst discs within the design specification limit pressures. The testing for burst discs is described in Section C.6.3.2C.

The shutdown heat removal function requires the integrity and operation of the steam piping, main steam line valves, turbine bypass valves, steam generator modules and steam drums. All of these components include levels of redundancy during shutdown heat removal. Light Water Reactor and conventional steam plant experience and data as well as acceptance tests will be used to assure that adequate SHRS reliability can be established without special testing directed toward these components.

C.4.2.4 Direct Heat Removal Service (DHRS)

Areas Identified for Reliability Emphasis

The DHRS incorporates two primary coolant flow paths. An inner loop transfers heat from the core to the outlet plenum via circulation in the PHTS. Heat rejection from the outlet plenum is accomplished via injection of cold sodium into the outlet plenum via the makeup nozzle and extraction of hot sodium via the overflow nozzle. An essential element for the successful operation of this system is the effective heat transfer between sodium circulating in the two paths. This heat transfer takes place by means of mixing of sodium from the two circulation loops in the outlet plenum. The effectiveness of this mixing mechanism has been demonstrated in the 1/21 scale water tests performed at ARD. Further confirmation has been obtained from the 1/4 scale water tests conducted in the Integral Reactor Flow Model at HEDL.

The DHRS uses the components of the primary sodium service system and the EVST cooling system. The DHRS introduces only the overflow heat exchanger and additional valves. The integrity of primary piping and other elements of the DHRS coolant boundary will be supported by the materials testing programs identified for the primary coolant boundary in Section C.6.1.2. The performance of DHRS will be supported by information from flow testing of the reactor vessel outlet plenum described in Section C.6.1.2A. Other testing includes performance testing of active pumps and valves; design verification testing of the air blast heat exchangers and manufacturer acceptance testing of the overflow heat exchanger and the air blast heat exchangers. The air blast heat exhangers are similar to the FFTF air blast heat exchangers and their reliability will be supported by testing done for the FFTF components.

C.4.2.5 Interfacing Systems

Areas Identified for Reliability Emphasis

The SHRS has the capability of functioning in the natural circulation mode in the primary, intermediate and steam/water loops. The requirement for electrical power is that the battery supply be available to operate control instrumentation in the SGAHRS. The components of the power supply are of conventional design, and generic reliability data are available to support their reliability.

C.5.0 Reactor Shutdown System Evaluation

A program element essential to meeting the objective of reliability enhancement is the timely feedback of data from the program activities to the plant equipment design, fabrication, installation and operation activities. In the case of analytical assessments, this is achieved by requiring that the reliability assessments be a part of the design support package for each component which is part of the RSS. Assuring timely feedback of data from the test program, however, requires careful planning since in many instances test articles cannot be made available before a number of the design and fabrication processes have been completed. In recognition of this problem, the schedule for the Reliability Program test activities has been coupled to that for the plant component design, fabrication, installation and operation activities. All test activities will provide data in advance of operation of the plant units. The testing schedule is such that positive response is possible for the elimination from the plant equipment of any unacceptable features uncovered in the test program.

The analysis of the RSS includes: (a) qualitative analyses (FMEAs and CCFAs) which identify potential random independent and common cause failures, (b) evaluations of failure consequences, (c) numerical reliability predictions of potential failure modes to supplement design analyses, (d) evaluation of test results to provide input to failure resolutions, (e) evaluations of design changes or updated details for impact on failure modes, (f) continuing evaluations of critical dimensions or processes through manufacturing and installation to minimize potential errors and (g) assessments of interfacing components potential failure modes and consequences.

RSS analysis utilizes the preliminary FMEA as a starting point for further analysis as well as test definition. Based on this FMEA, failures having common cause potential for scram failure of more than one control rod are identified. Priority is then given to the resolution of the common cause failures in both analysis and test efforts. Each failure mode is analyzed to determine design margins or design features which protect against the failure mode. Evaluations of test results are also factored into the failure mode analyses. If marginal or inadequate protection against the failure mode is indicated, the system level consequences of the failure are evaluated to determine need for additional protection. This process is used to assure acceptably low likelihood for the failure, to determine acceptable consequences of the failure or to identify design changes for reliability enhancement.

Initial FMEAs were utilized in the preliminary design reviews of RSS components. Periodic updates of the FMEAs reflect new analyses, test data, design improvements, etc. to show that failure modes are precluded or their effects are nullified. Each failure mode identified is being addressed to ensure that it will not impact RSS reliability. Material and process specifications and installation and operation procedures are being evaluated from a reliability viewpoint. Design changes, manufacturing waivers and nonconformances are also evaluated as appropriate to ensure RSS reliability.

C.5.1 Primary Mechanical Subsystem Evaluation

C.5.1.1 Analysis

Analytical models for the unlatching and scram insertion functions have been developed to assess PCRS reliability. The impact of design changes or more detailed component design features are assessed using these models. Included in PCRS evaluations are updates of control rod system misalignments resulting from interfacing component design changes or updates of reactor system installation details. Design changes not directly impacting unlatching or insertion analyses are assessed to assure negligible impact on the shutdown systems.

Development of the analytical model to predict pin lifetime behavior and scram performance characteristics is closely allied to the test program. Data from the control assembly tests, are vital to this model development which will be used in the reliability and design analyses.

The reliability analysis and test evaluations will be summarized in the PMS Reliability Design Support Document. FMEAs for each PCRS component are prepared for component design reviews. Updates of the component FMEAs are to be prepared to support significant component milestones (e.g., final design, test program completion). The PMS Reliability Design Support Documents will be prepared to encompass the entire, completed series of PMS reliability analyses and testing activities.

Table C.5-1 provides a summary of the principal PCRS scram failure modes identified from qualitative analyses. Actions to evaluate failure modes includes assessments of design features, testing and supporting analyses. These principal areas are detailed in Table C.5-1. Comments are given in the last column of the table to elaborate on the identified areas for failure mode resolution. Preliminary design analyses indicate acceptable scram performance for each of the identified areas. The reliability efforts are directed at resolution of uncertainties in the design analyses and experimental test confirmation of the design predictions.

Washout of B_4C from absorber pins under assumed failed cladding conditions has been evaluated. The predicted total loss of B_4C from one or two pins results in only a few percent loss in control rod reactivity worth. Washout test data for B_4C pellets exposed to flowing sodium indicated low B_4C loss rates. Therefore, loss of B_4C from pin failure is not considered to be a significant failure mode.

Numerical Assessments

Numerical analyses have been performed to determine (a) the unlatching performance for FFTF test units to obtain an indication of the PCRDM unlatching reliability, (b) the PCRDM time to unlatch and (c) the PCRS time to insert in order to assess the PCRS probability to meet design requirements for negative reactivity insertion. These analyses assist identification of potential problem areas for further design, reliability evaluation and testing emphasis. Summaries of these analyses are given below and overall conclusions are given in Section C.7.1.1.

A. Unlatching Performance for FFTF Test Units

The FFTF CRDM is essentially the same design as the CRBRP PCRDM except for minor sizing differences to meet CRBRP load requirements and small changes to the segment arm springs and leadscrew bushings to enhance the scram reliability. The failure modes challenged in the integral CRDM unlatching test included those associated with part failures, friction coefficients, galling, design or manufacturing errors, leadscrew chips and misalignments. On the basis of the design and manufacturing similarity between the FFTF and CRBRP CRDM's, the FFTF test data provides a valid indication of CRBRP unlatching reliability. No failures have been found in 3513 FFTF test scrams of the test unit. This compares with the 750 scram events included in duty cycle for the CRBRP CRDM.

B. Unlatching Performance for CRBRP PCRDM

An analytical unlatching model of a CRDM was developed to predict the CRBRP unlatching time and the standard deviation for the unlatching time. The analytical model includes variables associated with the stator field decay time, segment arm springs, friction coefficients and loading conditions of the CRDM and leadscrew.

To get an accurate representation of the stator field decay, FFTF motor test data were used along with FFTF Environmental Life Test data. These data were used to calibrate the stator current decay equation. A single decay equation was fitted to the mean of all test data for field decay. Standard deviations were obtained by analyzing the spread of test data compared to the mean curve. Variations in the field decay due to temperature and critical current (two or three phase operation) were encompassed by the standard deviation. This procedure led to a conservative standard deviation as the FFTF test variations in input current and stator coolant flow lead to a broader distribution than expected for fixed plant operating conditions for these variables.

Mean friction coefficients based on material couples tests were adjusted to improve agreement between calculation and test results for FFTF unlatching tests. Friction coefficient distributions were defined using data obtained from material couples tests. The model and data were then used to predict FFTF unlatching test results. Predicted unlatching times compared well with test results. The predicted standard deviation was, however, considerably larger than that obtained from testing.

The unlatching model was then updated to CRBRP PCRDM design parameters which included preliminary electric current decay data from PCRDM motor test data. This procedure changed the mean field decay curve but the standard deviations from FFTF tests were retained as a conservative envelope since the test results covered a broader range of operating conditions.

The PCRDM model was used to aid assessments of scram time failure modes associated with the springs, stator and CRDM friction. Analysis predicted a mean time to unlatch of 0.089 seconds with a standard deviation of 0.010 seconds. This unlatch time was combined with the scram insertion times (see Paragraph C below) to provide a comparison with overall scram time requirements. Since this analysis was completed, a design change was implemented to reduce stator wire diameters (increasing resistance) and thus decreasing the unlatching time.

C. Scram Insertion With Design Basis Misalignments

This analysis was performed to assess insertion reliability against potential failure modes associated with variability of misalignments within the design envelope, sliding friction coefficients, flow parameters and scram spring constants. Distributions were assigned to these variables which were then Monte Carlo sampled to perform probabilistic analysis. The scram spring force and misalignment distributions utilized for this analysis were based on the design specified tolerances for the PCRS and interface components. For individual parts, uniform probability distributions over the maximum drawing tolerances were assumed for each gap. This assumption of uniform distributions is conservative compared to asymmetrical gamma distributions (pealed towards smaller gap size) typically found for manufactured parts as it leads to greater probability at the extreme tolerance limits. Since most parts contributing to these misalignments have 100 percent dimensional inspection requirements, there is a very low probability of a part exceeding drawing tolerances.

These distributions are then combined for all parts, leading to the overall misalignment at a given elevation. The resulting distributions at a given elevation approaches a truncated normal distribution. The extreme tails of the distribution are, however, included in the analysis for added conservatism. The flow parameter distribution utilized uncertainties obtained from the FFTF control assembly flow test. This test was run in water with a prototypic control assembly. The data was then correlated to flowing sodium conditions. From this, a friction factor was derived. The percent error based about the mean value was used to define uncertainties. Friction coefficients and associated standard deviations used for gamma distributions were obtained from material couples friction and wear tests performed under Base Technology programs.

These analyses yield the probability of achieving the design requirements for scram insertion speed. Combining the unlatching time and insertion time analyses for the time from start of stator field decay to insertion of 1$ of reactivity yielded a mean time of 0.338 seconds and standard deviation of 0.012 seconds. These results indicate a normal scram probability of >0.999 (per challenge) satisfying the scram time requirements under operation within the design basis and no structural failures.

C.5.1.2 Testing

PCRS tests have been planned to determine possible design deficiencies and investigate postulated failure modes. Testing is maximized under those operating conditions postulated to cause failure, especially where it is desired to supplement current data to determine design margins against potential failure.

Manufacturing processes have been considered throughout the test program planning. Included in this planning are:

1) Plant unit specifications are used for all prototype procurements to assure resolution of potential fabrication problems. No prototype exceptions have been permitted for the PCRDM/PCRDs. For the prototype PCAs to be used in sodium loop testing, the only exceptions are non-prototypic pin internals (no B_4C) and changes to material standards (ASME standards substituted for RDT standards) for absorber pin cladding and minor non-wear limited parts. Fabrication, inspection, and acceptance test specifications are the same for prototype and plant units.

2) Simultaneous material procurements have been made for prototype and plant unit PCRDM/PCRDs. Potential plant unit failure resulting from material variability should be minimized as material deficiencies are expected to be identified in the prototype tests.

3) Acceptance tests for each unit will be performed by the PCRDM/PCRD vendor prior to shipment. These tests will include functional tests of the PCRDM to compare performance with acceptance requirements.

To minimize potential failures resulting from installation and operation, the following activities are planned:

1) Prototypic installation employing plant installation tools and procedures will be used throughout the PCRS test program.

2) Prototypic testing of all planned PCRS maintenance operations employing plant maintenance tools and procedures will be used to search for human factors or design errors which could lead to scram failure.

3) CRBRP startup tests will include functional and scram tests to verify shutdown performance prior to criticality and during power ascent.

4) CRBRP scram tests will be performed after every reactor refueling prior to approach to criticality.

5) Normal shutdowns will be completed by a scram test of control rods. After control rods are inserted sufficiently to shut down the reactor, the rods will be scrammed to complete insertion to test scram performance.

The following paragraphs identify the individual tests and discuss the engineering features of each test. The feedback to the plant equipment development program is identified together with the options available for responding to the test data. A description of the test facilities to implement these tests is included in Addendum 1.

A. <u>PCRS Prototype Design Test</u>

The PCRS Prototype Design Test includes four parts: the PCRDM Accelerated Unlatching Life Test, the PCRS Prototype Design Test, the Disconnect Actuating Tool (DAT) Test and the Maintenance Equipment Tests. In the unlatching test, the PCRDM will be operated beyond the design life of unlatch and travel to assure margins against wear related failures and to eliminate design defects. In this test, operating environment extremes such as mechanism misalignment, temperature and pressure will be increased beyond design basis conditions to evaluate design margins. The PCRS Prototype Design Test is a complete control rod system (PCRDM/PCRD/PCA) test in a sodium environment. PCRDM and PCRD performance data such as unlatching time and scram insertion time will be used to assure that design specifications are satisfied under design basis operating conditions of misalignment, sodium flow rates and temperatures. The DAT and Maintenance Equipment Tests provide data which will be used to thoroughly evaluate the maintenance procedures on prototypic equipment under plant type operating conditions. These tests are to identify weaknesses in the equipment design and the maintenance procedures as well as to evaluate any maintenance related failures.

Results from this test are available for the period of early 1978 through late 1980. The PCRDM/PCRD Final Design Review was held in October 1978, and data from the CRDM Accelerated Unlatching Life Test was available for this design review. By mid 1978, manufacture of all prototype PCRDM/PCRDs was completed. Fabrication of the plant units progressing in parallel with the testing. Both the test and plant units fabrication will be completed in late 1980. The overlap of testing and fabrication has permitted PCRDM/PCRD design changes, identified as desirable from the test program, to be incorporated into the plant units. Design changes based on test results have been made to facilitate installation and maintenance. Normal operation and safety-related performance has exceeded design requirements.

B. PCRS System Level Test

The System Level Test has elements concentrating on different aspects of PCRS performance.

Part I is the Real Time Test of a prototype PCRDM/PCRD and prototype PCAs. Representative hold times (inactive periods during which the control rod is not moved) are interspersed throughout the accelerated operations. The operating profiles of a Row 4 corner control rod are simulated at an accelerated rate of cyclic operation because rods at these positions are not used for daily power control and experience periods of inactivity during normal reactor operation. Besides providing additional data to assure manufacturing variations do not affect design margins against potential wear related failures, the hold times generate data to confirm that potential time related failure mechanisms such as self-welding are not significant. Scram times and other performance data are used to confirm that design specifications are satisfied and to assure the reliable operation throughout the test.

Part II is the Failed Bellows Test and consists of operating a prototype PCRS for one year with an intentionally failed bellows to determine potential related failure modes for PCRDM unlatching and PCRS insertion. Bellows failure will expose parts normally in an argon environment to sodium vapor. Scram release time and wear will be monitored to evaluate design performance and margins under failed bellows conditions. By observing areas of sodium buildup or extreme wear, potential failure mechanisms resulting from a failed bellows will be identified.

The PCRS flow vibration test, Part III of the system level test program, utilizes accerlerometers on the PCRD and shroud tube in the area of the dashpot cup and on the PCA outer duct to monitor flow vibration effects. These data, together with past sodium test examinations of all test components, are used to verify acceptability of the PCRS design relative to flow vibration effects.

Results from these tests will be available so that any plant unit modifications can be implemented prior to their shipment to the site. All reliability testing will be completed prior to initial startup testing in CRBRP. Test results can be factored into the PCA final design, scheduled for completion in late 1981.

The PCRS System Level Test facilities have been designed for testing at the extremes of the design operating conditions and beyond to induce failures and thus determine design margins to assure reliable performance. Maintenance equipment including a simulated maintenance pit will be used in the system level tests.

C. PCRS Dynamic Seismic Friction Test

This test provides two pieces of information essential to the accurate prediction of control rod scram insertion performance during a seismic event. These are (a) the effect of fluid coupling on the lateral translational behavior of a driveline and control assembly within their respective guide members and (b) the effective coefficient of friction between the interacting components under conditions of short duration contact. Effect (a) is of importance because it dictates the number and magnitude of the lateral impulsive forces generated as seismic excitation causes the driveline and control assembly to "rattle" within their guide members. The frictional component of these loads acts to retard scram insertion, hence their number and magnitude reflects directly on the seismic scram insertion prediction. Effect (b) must be evaluated in order to reduce present conservatism in the friction assumptions used to convert the lateral impulsive loads into axial loads opposing scram insertion. During the brief period of lateral impact loading, it is possible that squeeze film sodium lubrication will decrease the effective friction coefficient.

The test provides data on the translational behavior and impact load behavior of simulated rod/guide tube features when subjected to seismic excitation in a fluid environment. The impact load-time histories obtained are used to calibrate analytical models to assure the correct representation of entrained fluid effects. Drop times are to be measured. These data, together with the impact load-time histories has been used to determine the effective coefficient of friction under squeeze film lubrication conditions.

Results from this test were obtained in the period early 1977 through early 1980. Most of the test results were available prior to the PCRDM/PCRD Final Design Review in 1978. Friction coefficients from the test have been combined with normal impact forces from seismic analyses to show that PCRS seismic scram speed requirements are satisfied.

D. PCRS Friction Couples Test

Data will be generated by these tests is used to evaluate friction and resultant drag forces that will be encountered during PCRS operation. The materials used in the PCRS design have been carefully selected, especially in the areas where contact during operation is anticipated. These tests provide friction data on the material couples under varying conditions of contact force, temperature, environment (liquid sodium, argon-sodium vapor and argon), length of contact surface and time between operations. The maximum friction developed under these conditions has been incorporated into scram speed analyses an analytical model to confirm design margins.

The test material samples consist of a pin and plate. These samples are placed in a facility capable of providing reciprocating motion and recording friction over the range of conditions specified.

Results from these tests are currently available. These data are utilized available for both the PCRDM/PCRD and the PCA Final Designs.

E. Control Assembly Hydraulic Test (Flow Test)

This test generates flow, vibration and pressure drop data to characterize the hydraulic performance of a prototype PCA. These data are required to assure that adequate design margin against control rod flotation is available. Flow induced vibration will also be investigated to check the rod bundle response to flow turbulence up to 150 percent of nominal flow. The test facility will be a circulating water loop with the required flow and pressure drop instrumentation.

Results from this test are being utilized in the PCA final design evaluations.

F. Control Assembly Rotational Joint Test

The purpose of the Rotational Joint Test was to verify the performance of the rotational joint under expected operating environments. The objective of the test was to measure the torque transmitted through the joint under prototypic temperatures and loads. In addition, the effect of scram impact dynamic loads, misalignment of input and output shafts and sodium soak were determined. Finally, the effectiveness of the rotational joint in reducing duct wear was determined by purposely inducing wear pad to duct contact and cycling the rod until approximately six times the goal lifetime travel is achieved. Wear pad to duct contact was reestablished after every half lifetime of travel. Data generated by these tests demonstrated the effectiveness of the rotational joint to minimize control assembly wear.

G. Primary Control Assembly Pin Bundle Compaction Test

Data generated by this test is used to calibrate the pin bowing analysis. Pre-bowed pins were compressed to the configuration required by the control rod inner duct. The forces necessary to compact the pins to the bundle dimensions will be measured and recorded to determine pin contact loads with the duct and with other pins. These data are combined with analysis to establish potential outward deformation of the control rod inner duct as a result of forces due to pin bowing. Results from this test are available for incorporation into final PCA design efforts.

H. Control Assembly - Drag Force for Bowed Duct Test

Control rod duct bowing resulting from irradiation and thermal gradients is a potential common cause failure. To assure that adequate design margins exist to eliminate this source of failure, drag load measurements during insertion and withdrawal were made under various bow conditions and environments. Prototype ducts were bowed in a test facility where the bow, inner to outer duct orientation, flow rates and radial misalignments between a simulated lower driveline and the outer duct were varied and recorded. The bow configurations that can cause insertion failure due to excessive drag forces were determined and the margin between failure and worst-case design conditions were established. This test also showed that the three dimensional mechanical/structural interactions between the driveline/control rod and associated bushing/outer duct under misaligned conditions can be adequately evaluated by two dimensional analyses. Effects of the rotational joint in the control rod shaft (which reduces both lateral and rotational contact loads between the control rod and outer duct) were included in the measured drag forces. Measured drag forces from this test have shown that duct bowing causes negligible drag forces for duct bowing exceeding worst-case design predictions. Bowing does not induce large retarding forces until the design criteria limit of forced three point contact between the control rod and outer duct is exceeded.

Results from this test will be produced from early 1977 through early 1978. Data will be incorporated into final PCA design efforts.

I. PCRS Seismic Test

To provide data that will confirm design margins against scram failure during an OBE or SSE, a prototype PCRS will be mounted in a test fixture coupled to eight vibration generators in a water environment simulating sodium levels. The unlatch time will be measured and recorded with various vibratory inputs. The PCRDM will be mounted on a three-dimensional shaker table to evaluate the unlatching performance. The shaker table, together with an additional five lateral shakers, are planned for evaluation of scram insertion performance. The data will be assessed to discover design deficiencies and establish design margins. This test is primarily oriented toward providing unlatching and scram insertion data for verification of analysis methods. Sinusoidal inputs typical of the acceleration levels under OBE and SSE conditions will be utilized for these tests. Results from this test will be available by late 1979. At this time, the plant unit PCRDM/PCRDs will have been completed and will be ready for shipment. Any need for modifications can delay shipment since site installation does not occur until late 1981.

J. Pin Rupture Test

Pressure pulses from absorber pin rupture could result in sufficient inner control rod duct deformation to cause a scram failure. Data generated by this test are combined with analysis to confirm design margins are adequate against this postulated failure mechanism. Pins at different locations in the pin bundle were intentionally faulted and ruptured in prototypic ducts to obtain data such as pressure pulse magnitude, duration, pin plenum pressure decay and duct deformation. The resulting duct deformations have been found to be small and pin ruptures have negligible potential for causing a scram failure.

K. Duct Impact Test

A PCRS scram is terminated by impact of the driveline and scram arrest flange at velocities less than 14 inches/sec with the scram arrest flange welded to the PCA outer duct. Since the PCA duct loses ductility from irradiation, impact tests on irradiated ducts are planned to assess the potential for brittle fracture due to scram impact loads. Impact tests in support of the FFTF program were performed without failure on an irradiated EBR-II control rod thimble at ambient temperature conditions (75°F). Tensile data generated at temperatures (1000°F-1400°F) higher than the irradiation temperature (700°F) have shown a reduction in ductility which can be lower than the ambient temperature ductility. Since CRBRP control rod ducts can be impacted at up to 1000°F during scram operation after being irradiated at lower temperatures, impact test data are required to confirm scram impact acceptability.

The Duct Impact Test simulated scram impact by dropping known weights from varying heights on an irradiated EBR-II duct. The test included impact loads resulting in stresses well in excess of stress conditions expected in PCA ducts. Tensile test data from the ducts were also obtained to assist post analysis of the test and extrapolation to PCA conditions. This test will be used to define design margins against PCA duct failure due to scram impact. Results from this test are currently available and are being used to evaluate PCA design impact.

L. Duct Crushing Test

The purpose of the Duct Crushing Test is to investigate the failure mode of a highly irradiated hexagonal duct segment when subjected to lateral loading similar to that experienced by the above core load pads on CRBRP during seismic excitation. Material used in these tests is taken from EBR-II control rod thimbles previously irradiated to a fluence of approximately 1×10^{23} total fluence and between 4 and 5.7×10^{22} fast fluence. The material is in the form of hexagonal duct sections, similar in profile to the CRBRP core duct profile. The test material therefore incorporates the features which introduce uncertainty into the duct crush strength analysis. These are (a) a much reduced ductility with the attendant potential for brittle fracture, (b) strain concentrations at the duct corners and (c) plane strain bending stresses (the available ductility data on irradiated stainless steels have been obtained from tests in which the stresses were uniformly tensile).

Sections of an irradiated EBR-II SS304 duct were loaded in a transverse direction between two jaws to simulate in-service seismic loading. In addition, tensile and bending test specimens were machined from the duct to provide basic materials data for use in analytical predictions of duct response to transverse loading for subsequent comparison with test data. Temperature and strain rate were varied over a range consistent with expected CRBRP conditions to determine if any combination of these parameters would lead to a brittle fracture. Test temperatures were chosen to be higher than the average irradiation temperature of the duct, since the results of the prior EBR-II duct evaluation indicated a decrease in material ductility with an increase in test temperature above the irradiation temperature. Results from this test are currently available and are being used to evaluate PCA duct design.

C.5.2 Secondary Mechanical Subsystem Evaluation

C.5.2.1 Analysis

A summary of the principal SCRS scram failure modes identified from qualitative analyses is provided in Table C.5-2. The areas of testing, design features to mitigate consequences or prevent the failure, and supporting analyses which are important to failure mode resolution are referenced in Table C.5-2. Identification and evaluation of these failure modes have provided guidance for the appropriate corrective or preventive actions to minimize the impact on SCRS scram function. Further efforts are directed at resolution of uncertainties in the design analyses and at experimental test calibration of the design prediction methods.

Numerical Assessments

Available data indicate that the frequency of spurious scrams is highest at the beginning of operation of a reactor and decreases thereafter because of a learning process. The number of scrams which a reactor will see through its lifetime can therefore be estimated by the use of a mathematical model which takes into account this learning process. The current reliability assessment of the SCRS design using this model is that the design is adequate in terms of safety-related reliability.

C.5.2.2 Testing

The testing of the SCRS and its components is orientated to design verification; i.e., a determination of the capability of the design to meet its functional requirements. Data resulting from the design verification tests will also be analyzed from a reliability viewpoint, and reliability deductions will be made as the data permits.

The following paragraphs identify the individual tests and discuss the engineering features of each test. A description of test facilities for these tests is included in Addendum 1.

A. Latch Real Time Test

This test permitted evaluation of self-welding in this critical component early in the development cycle. Environmental conditions for the test were more severe than those predicted for the latch in reactor service.

The test articles were subjected to less vibration, a constant force, a higher and more stable sodium temperature and a longer time between scrams than will occur in the reactor environment. This provided accelerated testing of the potential for the self-weld mechanism. This test allowed the latch test units to remain dormant in the latched condition for a full year. A dead weight was hung from the latch to simulate the gravitational and hydraulic loads of full power operation. To achieve a baseline for assessing the impact of the dormant period, friction coefficient were determined prior to the start of the dormant period.

This test was successfully completed in January, 1980. There was no evidence of self-welding or bonding. From the initial evaluation of the results of this test, including a range of coefficients of friction value, it can be inferred that the SCRS latch system of the configuration and materials tested will unlatch in a prototypic environment after prolonged exposure to high purity, high temperature sodium.

B. Latch Scram Test

The Latch Scram Test demonstrated the performance of the latch assembly under normal and overstress operating conditions. The test also determined the extent of wear between the contacting surfaces of the latch assembly as a function of the number of operating scram cycles. Two latch units were tested in the final configuration in the test, and other units are being tested in the system test. Data from this latch test did not identify any latch failure modes and established the latch cyclic life capability as being well beyond the design life.

Latch/collet assemblies were tested in liquid sodium with operating temperatures ranging from 400°F to 1050°F. Each test unit was examined before and after testing to assess the condition and degree of degradation. During the tests, all externally monitored parameters were checked for out-of-limit conditions to provide a continuous assessment of test rig and latch performance. Latch scram test #1 was successfully completed in early August 1979, after being subjected to 1987 total scram cycles, a number equivalent to approximately five times the latch service life. Analysis of test data, primarily coefficient of friction values, indicated no significant effect on latch performance of wear due to repeated scram cycles. Distributional characteristics of the data when compared with the specified coefficient of friction limits were favorable. Post-test examination of the parts did not reveal significant wear at any of the critical interfaces.

Latch scram test #2, using a different test unit, was successfully completed in mid-September 1979, after accumulating 3795 total scram releases. This number corresponds to approximately ten times the latch service life. Nothing observed during the course of testing, or as a result of analysis of the coefficient of friction data indicates any significant affect of wear on scram performance. Latch release occurred in all cases within the lower third of the specified range for coefficient of friction.

C. Driveline Lower Bellows Test

The primary objectives of these component level tests were to assess bellows design adequacy and to obtain information on component life in a prototypic environment. The tests reproduced the bellows motion involved in scram actuation and recoupling. Test items were fully prototypic of the CRBRP SCRS design in all aspects, including configuration, material construction, dimensions and clearances.

Two sets of bellows were each tested in three phases. Phase I simulated refueling conditions, Phase II simulated full power conditions, and Phase III was the life test. Each set of bellows was cycled more than 3600 times, equivalent to ten times the design life. Based on the successful tests of both items it has been inferred that the bellows design is adequate and that lifetime characteristics are satisfactory.

D. Pneumatic Valve/Cylinder Test

Both cyclic and real time failure mechanisms are being evaluated in a prototypic environment in this test. Units tested are cycled to several times the design life or to failure, whichever comes first to provide failure information and prototypic component performance data.

Pneumatic Valve/Cylinder Assemblies are being tested in two phases as follows:

1. Cyclic Testing - One assembly was cycled at approximately two-hour intervals until 900 cycles (five design service lifetimes) were completed. A cycle consisted of ten consecutive poppet valve checkout cycles followed by a scram cycle. Scram time, valve poppet opening times, cylinder leak rate, and valve temperatures were recorded at periodic intervals.

   This test of Valve/Cylinder #1 was successfully completed in March 1980. The test objectives were achieved and no safety-related failures were encountered. Nothing observed during the course of the test would have affected the ability of the valve to operate reliably from a safety (scram) viewpoint in a prototypic environment. Reliability analysis of the test data indicated adequate operating and design life margins. Analyses of valve/cylinder scram time to provide distributional characteristics showed a high probability of the valve to support the SCRS scram time requirement.

2. Real-Time Testing - A second assembly will be held in the operating mode for about 12 months. At the end of the hold period, the test article will be test cycled 900 times or to failure, whichever occurs first. The valve poppet opening time, cylinder leak rate, and temperature will be recorded at intervals during the cyclic portion of the test. Scram time at the end of the operational hold testing will be measured. Upon conclusion of the test, the resulting data will be analyzed and conclusions drawn regarding operating and design life margins, standby reliability and performance reliability (i.e., the ability to meet specified scram time requirements). Checkout of this assembly commenced in October 1980.

E. SCRS Failed Bellows-Extended Limits Test

This test is intended to evaluate the capability of the design to meet its functional requirements for periods up to 11 months under adverse operating conditions associated with failed bellows. The main shaft bellows and the driveline lower bellows protect the Secondary Control Rod Drive Mechanism (SCRDM) and Secondary Control Rod Driveline (SCRD) from sodium vapor. The major concern resulting from a bellows failure is exposure of the SCRDM and SCRD internals to sodium vapor. Condensed sodium vapor between close-fitting moving parts could result in potential interferences which could, in turn, cause degradation in the performance of the latch release action. This test will be run with both the Mainshaft Bellows and the Driveline Lower Bellows deliberately faulted to simulate the expected mode and magnitude of bellows failure. The test will demonstrate the extent to which sodium vapor can diffuse through the argon cover gas, onto the surface of moving parts of the SCRDM and SCRD, and the degree to which performance may be degraded.

Except for the purposely damaged bellows, the test article will be of the plant unit design.

Testing to be performed includes characterization testing at various sodium flow rates and temperatures, system hold and scram testing, motor test, position indication test, LVDT displacement test, and pneumatic scram valve poppet movement test. Analyses of test data will be made to draw inferences concerning safety, design margin, and scram performance. The test data will also contribute to reliability assessment of the pneumatic valve/cylinder and the latch.

F. Bowed Guide Tube Bowed Test

This test will determine the amount of deformation that the guide tube can accommodate without adversely affecting scram time. Distortion of the guide tube beyond the design limit could degrade or prevent insertion of the control rod after unlatching is completed. This test will provide data regarding scram times where a control rod is interacting with a deformed guide tube. Water will be employed as the testing fluid. The guide tube bow will be incrementally increased until control rod insertion is prevented or substantially affected. A scram will be performed for each distortion increment. The hydraulic assist force, the water temperature and the argon pressures will be monitored during the test. The scram time, guide tube deformation, degree of insertion of the control rod, and the guide tube and control rod dimensions will be recorded. The results from this test will be produced in 1981 and 1982. Scram time data will be analyzed to assess to probability of exceeding maximum allowable scram times versus a given degree of bowing.

G. SCRS Prototype-1 Test

The first SCRS prototype system test (P1) was successfully completed in December 1978. The objectives of this test were to provide a proof-of-principle demonstration of the design, to identify operating characteristics and provide a basis for assessing operating margins, and to expose failure mechanisms that had not previously been predicted. The test was carried out over a wide range of temperature and flow conditions, both above and below the anticipated operating range. The Prototype 1 test article successfully completed 1570 full scram insertions, which is more than twice the 700 scrams expected of the SCRDM over the 30 year plant life.

At the end of these extensive tests, Prototype 1 was still performing within specification requirements. No safety related failure modes occurred during testing, and there was no evidence of incipient failure encountered upon post-test disassembly and inspection. The test data and the post test observations, therefore, support the conclusion that the design is sound and incorporates adequate margins for the intended use of SCRS.

The P1 test results identified several areas in which design improvements could be made to enhance fabricability, maintainability, and performance. These changes, as well as others, were included in the Prototype 2 test article.

H. SCRS Prototype-2 Test

The major objectives of the second system test, SCRS Prototype, are to verify the ability of the SCRS design to meet its functional design requirements under expected operating conditions, to identify operating margins, to evaluate design improvements incorporated as a result of the P-1 experience. Testing will be performed to verify satisfactory operating under prototypic conditions, and to determine sensitivity to variations in such operating parameters as sodium flow and temperature, control rod elevation, misalignment and scram cylinder pressure. Repetitive scram cycles will be conducted at various combinations of these parameters. Hold testing will maintain the SCRS in the ready-to-scram position at combinations of sodium flow rate and temperature of 10%/400°F and 110%/1050°F. A series of scrams will be performed before and after each hold period. Throughout the test, the pneumatic scram valve will be periodically subjected to poppet movement tests.

Data from the P-2 test will be analyzed for inferences pertinent to overall scram reliability, reliability of safety-related P-2 design changes, safety-related design margins and operating margins. Data from this test will also contribute to evaluation of critical components such as the pneumatic valve/cylinder, the latch and the bellows.

I. SCRS Prototype-3 Test

The objectives of the Prototype 3 (P-3) test are to verify the ability of the design to meet functional design requirements under design operating conditions, to identify operating margins by testing in excess of normal design operations, to expose potential failure modes which may not have been previously predicted, to evaluate the cyclic failure mechanisms in a prototypic environment and to demonstrate the ability to perform required maintenance operations.

During the course of this test, repetitive scram cycles will be conducted at a variety of sodium flow rates, sodium temperatures, and misalignments. The effect on scram performance due to these variations as well as changes in control rod elevation and pneumatic cylinder pressure will be determined. Hold testing will maintain the SCRS in the ready-to-scram position for combinations of sodium flow rate and temperature of 10%/400°F and 110%/1050°F. A series of scram cycles will be performed before and after each hold period.

Throughout the course of the test, the pneumatic scram valve will periodically undergo poppet movement tests.

The test data will be analyzed to provide inferences regarding scram capability, operating margins, and design margins. Test data will also contribute to an assessment of the pneumatic valve/cylinder, the latch, the bellows, and other safety-related components.

J. SCRS Prototype-4 Test

The objectives of the Prototype-4 (P4) test are similar to those given for P-3; This test, however, is the final system test prior to operation of the plant units and it is intended as the final checkout for the system and to demonstrate the ability to perform required maintenance operations.

The P-4 test article will be scram cycled so that all components undergo a number of scrams greater than their design service life. Testing will be performed to determine the system performance sensitivity to variations in operating parameters. The unit will also be held in the parked position for 11 months at prototypic full power conditions to expose passive-state failure modes and mechanisms.

Data from this test will be analyzed for inferences concerning system scram capability, standby reliability, design margins, and operating margins. This test data will also contribute to a reliability evaluation of the pneumatic valve/cylinder, the latch, the bellows, and any other safety related components.

C.5.3 Electrical Subsystem Evaluation

C.5.3.1 Analysis

To supplement a system level FMEA, qualitative and quantitative reliability analyses are performed on each module in the Electrical Subsystem. The qualitative analysis consists of an FMEA at the piece part level which considers identifiable failure modes of the piece parts. This analysis lists assumptions made during the analysis such as piece part failure state and the effect of the assumed failure. The FMEAs will be updated as needed to document the current status of the design.

The qualitative analysis also considers the effects of the assumed failure on other piece/parts in the circuit and whether the assumed failure has the potential to cause additional part failures or overstress conditions in the circuit and whether these failures would be safe or unsafe. The quantitative analysis, using part stress analysis techniques, is performed on a module basis. A reliability prediction of each module is being made using MIL-HDBK-217B or other data sources as appropriate. The information from the FMEA is then used in conjunction with quantitative analysis to predict the unsafe failure rate of each module.

Numerical Assessment

A current numerical assessment documented in Reference 2 includes a quantitative evaluation of the primary and secondary electrical subsystems in relation to their ability to function.

A model was developed to evaluate the reliability of the primary and secondary subsystems as they functioned under a specified set of plant operating conditions and procedures. Input data to the model consisted of component failure rates, test intervals and other parameters characteristic of ES operation. Failure rate data used was based on either detailed predictions using MIL-HDBK-217B or other reliability studies conducted for the FFTF program which are appropriate for CRBRP equipment. Other model input parameters were based on planned operating procedures.

Numerical assessments have been conducted at both the module and system level. Results from this analysis indicate that the ES is not a significant contributor to the safety-related unreliability of the plant. Data obtained from the ES test program will provide further support for the failure rates used in this assessment.

C.5.3.2 Testing

The test program for the ES equipment is made up of two basic types of tests: qualification tests and extended operations tests. Qualification tests will be performed by the vendor primarily at his facility. Qualification tests provide evidence that the as-built equipment meets the requirements of the procurement specification. Extended operations tests will be performed. These tests provide a means by which extensive operating experience can be accumulated, resulting in both reliability growth and reliability demonstration. Reliability growth results from identifying and correcting any design, fabrication or maintenance weaknesses before the equipment is installed in CRBRP.

A. Qualification Tests

The qualification tests can be classified as preproduction, production or acceptance tests. These tests are described below:

1) Preproduction Tests

Prototype modules undergo a series of tests to verify that the design meets all the requirements of the procurement specification.

The preproduction tests are implemented by first testing each prototype module so that a set of baseline data can be developed. Later test data are compared with these baseline data so that any degradation can be detected.

Each prototype is then subjected to thermal conditioning to detect any failures due to design, fabrication or workmanship problems. During this thermal conditioning, each prototype module will be subjected to 10 thermal cycles in which the temperature is varied from $-30^{\circ}F$ to $150^{\circ}F$ at rates between $9^{\circ}F$/minute and $30^{\circ}F$/minute. The temperature is held at the high and low extremes for a minimum of 30 minutes with power applied to the modules for intervals over this range. The modules are then baked at $150^{\circ}F$ for 200 hours. These test conditions are substantially more severe than the specific design conditions for the modules.

After thermal conditioning, each prototype module will undergo functional and performance checks while subjected to worst case environments including temperature, humidity, power supply voltage and frequency, electrical noise and vibration.

These tests were completed in early 1977. Design and component changes required as a result of the prototype preproduction tests were factored into the manufacture of the production units.

2) Production Tests

After the project was satisfied that the design and manufacture of the prototype modules met all functional, performance, quality and reliability requirements, the production modules were manufactured. The production modules include plant equipment, spare equipment and equipment to be used in the extended operations test.

Each production module underwent a thermal screen consisting of a 36 hour period of power off, temperature cycles between the limits of $-4^{\circ}F$ and $185^{\circ}F$. Each module was then subjected to full functional and performance testing to verify that each module meets its requirements.

These tests were completed in mid 1977 in the case of the reliability units and in mid 1978 in the case of the plant units.

3) Acceptance Tests

The plant equipment undergoes acceptance tests in addition to the production tests. In acceptance testing, the modules are installed in their respective panels and the complete system wired together. A full set tests which verify wiring insulation strength. The equipment will be operated in this configuration for a minimum of 125 hours.

These tests were completed by the vendor in early 1980.

B. Extended Operations Tests

Extended operations tests will be performed. For these tests, the modules are connected to form a complete electrical system. Additional modules are also interconnected to simulate subsystems of the electrical system, such as additional logic trains. Configuring the modules in this manner allows data on long-term effects of operations on performance parameters to be collected. These data can be used to determine calibration and test periods and will be factored into the plant operating procedures. These long term performance measurements provide additional supporting data to confirm that the performance characteristics and propagation delays assumed in the analysis are conservative.

Maintainability information is being generated on these prototypic system configurations and can be used to confirm maintenance design plans and also as a basis for preparing maintenance procedures. Maintenance and calibration procedures from the vendor supplied manual will be followed, where appropriate, to provide assurance of their validity. Also, trouble shooting procedures from the manual will be followed when failures are detected. Problems detected from use of these procedures will be factored into the preparation of the plant operations manual.

Functional and performance tests, as listed in Table C.5-3, will be performed on the primary and secondary subsystem components. The purpose is to determine whether they complete their intended function when called upon to do so and to check if the function is completed within specified time limits. The functional tests consist of providing voltage pulses or switch closures, as appropriate, at the inputs of the test components and checking the response from the appropriate outputs. The performance test includes measurement of propagation delay. This is done by inserting voltage pulses at the inputs and checking the response of the test systems.

As a minimum, functional pulse testing will be completed on each test component once a shift. The functional tests which require input from an operator (e.g., manual trip, bypass instatement) will be performed once a week. The flux signal transmitters will be checked for signal propagation.

The propagation delay tests are performed once a shift in conjunction with the component functional tests. The propagation delay of the breaker is tested and recorded weekly. Performance tests are completed once a week on all components except the flux drawers which are checked daily. The frequency of the functional tests will be increased if the environmental parameters drift beyond specified limits.

Test Failure Reporting, Analysis and Corrective Action

A closed loop failure reporting and corrective action system has been implemented to assure that any hardware reliability problems encountered are corrected and to force reliability growth. Failures and discrepancies occurring are documented in failure/discrepancy reports. Reliability Engineering is the focal point for the failure reporting and corrective action system. The failures reported will be screened and failure analysis performed, as appropriate, to identify underlying failure mechanisms. Each identified failure mechanism will be evaluated to assess the need for corrective action and the type of correction action required.

C.5.4 Interfacing Components Evaluation

C.5.4.1 Analysis

A Reliability Design Support Document will include assessment of the failure effects of all of the RSS interfacing components and systems that appear on the Reliability Related Components List. The interfacing component assessments include FMEA's and resolution of the failure modes through design margins and system features limiting the consequences. Shutdown system performance evaluations will determine the consequences of potential interfacing component failures. Since the interfacing component failures are potential causative factors for common cause failures of the shutdown systems, interface component assessments will be given high priority. The initial reliability reports will be completed, reviewed and updated (as applicable) prior to the components' design reviews.

Through the FMEA's for interfacing components, failure modes have been identified which have the potential to degrade the combined PCRS and SCRS insertion function. Examples of these include:

1) Large and/or intermediate plug rotation with rods withdrawn

2) Secondary control assembly flow starvation

3) Upper internals structure sheds fragments from thermal striping effects

Each of these failure modes has been assessed and has associated corrective or preventive actions to preclude adverse impact on combined PCRS-SCRS reliability. The following presents examples of the results of the assessments.

Postulated rotation of either the large or intermediate rotating plugs results in misalignment of both PCRS and SCRS. Several degrees of rotation may be sufficient to influence PCRS insertion. The SCRS, being less susceptible to misalignment, requires a larger amount of plug rotation to prevent insertion. Action relevant to this failure mode consists of the incorporation of a series of mechanical locks installed prior to reactor operation and designed to resist all forces that could conceivably cause rotation including motor torque.

Hydraulic assist is used in the SCRS to accelerate the control rod downward during a scram. While the control rod will insert without the hydraulic assist, its insertion time is extended. To assure that the SCRS always operates at maximum efficiency, it is necessary to assume that the design sodium flow is available at the SCA nozzle during power operation. The required flow is assured by means of features incorporated in the design of the core support structure. Flow blockage prevention is achieved by a combination of debris barriers and auxiliary flow ports. A description of the flow blockage prevention features is provided in Section 4.2 of the PSAR.

Fragmentation of the metal surfaces of the upper internals structure could be caused by thermal striping. Metal fragments could become lodged in any control assembly duct and adversely affect the rod's ability to insert. Actions relative to this failure mode included a design change from stainless steel to Inconel 718 for upper internals structure component parts. Items such as instrument posts, chimneys and shroud tubes exposed to thermal striping conditions are being made of Inconel 718. Analyses of the upper internals have shown that margins against this failure mode are now adequate.

Each interfacing component will be analyzed in a reliability assessment as described in Section C.1.3.2. Failure modes described above would be addressed in reports associated with the reactor closure head, core support structure and upper internals structure, respectively.

TABLE C.5-1 PCRS FAILURE MODES AND RESOLUTION SUMMARY

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
PCRDM	Excessive retarding forces prevent or slow unlatching	a) Excessive friction or wear	Increased design margin	Life	Unlatching Model	Increased segment arm spring force margin
		b) Failed bellows	Purge gas	Failed Bellows		Purge gas minimizes sodium vapor in PCRDM
			Increased bellows convolutions			Bellows stress and failures reduced by increased convolutions
		c) Misalignment	Rotating plug locks		Unlatching Model	Rotating plug locks prevent accidental plug rotation
		d) Installation and Maintenance Errors		Maintenance Procedures		Maintenance tools and procedures tested in system level tests
		e) Magnetized Components		Real Time Life	Post Test Inspection	Magnetization can be checked by its effect on unlatch time
	Part failures prevent unlatching	a) Wear		Life		Testing exceeds required wear life for each CRDM
		b) Seismic	Seismic support	PCRS Seismic	Margin Analysis	Shield and Seismic Support Structure limits lateral deflection
		c) Manufacturing Errors		Prototype Unit Testing	Manufacturing	Testing of units from prototype and plant unit manufacturing sequences to identify potential manufacturing errors
PCRD	Excessive retarding forces prevent or slow insertion	a) Excessive friction or wear	Large clearances	Life	Insertion Models	Life tests exceed required wear life
		b) Seismic		Dynamic Friction	Margin Analyses	Shaker tests to obtain friction data and to calibrate seismic insertion analyses
				PCRS Seismic		
		c) Misalignment	UIS key lateral restraints	Prototype Testing	Insertion Models	Misalignment test

TABLE C.5-1 (Cont'd)

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
PCRD	Jamming of dash-pot cup or piston	a) Foreign parti-cles	Startup filters	Life		Special core assemblies have filters for initial sodium cleanup
			Inlet module features			Inlet modules provide debris barriers and strainers
		b) Galling		Life	Test Evaluations	
		c) Flow induced vibration de-formation	Shrouded PCRD	PCRS Flow Vibration	Test Evaluations	PCRS sodium loop tests provides an extended test period to evaluate vibration effects. Vibration measured in PCRS Flow vibration test phase
				IRFM		Integral Reactor Flow Model provides vibration data
PCA	Duct deformation retards or pre-vents insertion	a) Irradiation induced bow-ing	Increased clearances	Duct Bowing	Bowing Margins	Duct Bowing test to establish drag forces and failure point for varying duct bows
		b) Scram impact on irradiated duct		Duct Impact	Impact Evaluation	Completed duct impact test shows no failure even at im-pact loads in excess of de-sign values
		c) Pressure pulse from pin fail-ure distorting inner duct	Design for no failures	Pin Rupture	Test Evaluation	Pin rupture test to establish inner duct deformation for postulated pin failures
		d) Seismic loads on outer duct load pads	Heavy duct wall at pads	Duct Crushing	Crushing Margins	Test provides data support for analyses of all core assemblies
		e) Swelling and bowing of pins deforms inner duct	Pin to duct clearances	Pin Compact-ion	Design and Test Analyses	Test to correlate analyses for pin interactions and bundle compressibility
		f) Weld failure due to improper weld		Life, FFTF Irradiation	Post-test Inspection	Quality control during welding to prevent poor welds

TABLE C.5-1 (Cont'd)

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
PCA	Excessive retarding forces prevent or retard insertion	a) Galling or wear of outer duct	Incorporated rotational joint	Rotational Joint and Life	Post-test Evaluation	Added rotational joint with supporting test to minimize wear pad to outer duct loads
		b) Seismic		Dynamic Friction PCRS Seismic	Margin Analyses	Shaker test to obtain friction data and to calibrate seismic insertions analysis
		c) Foreign particles	Startup filters and inlet module features	Life		Life tests will simulate effects of design basis sodium impurities such as oxygen content
		d) Misalignment	Passive core restraint	Prototype Testing	Insertion Models	Passive core restraint eliminates potential for inadvertent errors in core restraint adjustments

TABLE C.5-2 SCRS FAILURE MODES AND RESOLUTION SUMMARY

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
SCRDM	Malfunction of scram valves or pneumatic actuator slows unlatching.	a) Foreign material	Cylinder piston bellows seal	Valve/cylinder	Post test inspection	Valve/cylinder will be tested beyond design life.
			Redundancy: 3 out of 5 poppets to Scram	SCRS Prototype		Valve design includes testable feature for in plant online checkout.
		b) Part failures	Redundancy: 3 out of 5 poppets required to Scram	Valve/cylinder, SCRS Prototype,	Post test inspection	Testing of component beyond design life will identify potential failure modes.
		c) Manufacturing errors		Valve/cylinder	Post Test inspection	Testing of components and system will identify potential manufacturing errors.
				SCRS Prototype		
SCRDM/SCRD	Excessive retarding forces slow unlatching.	a) Thermal effects	Large clearances to accommodate thermal effects	SCRS Prototype	SCRS structural analysis	Testing of units at prototypic temperatures to support analysis
		b) Argon contamination	Filter	Valve/Cylinder	Post test inspection	
			Buffer gas	Failed-Bellows		
		c) Excessive friction from wear, galling	Hardened wear surfaces	SCRS Prototype	Post test inspection	Testing of SCRS units will identify potential wear and galling.
			High actuation forces			Actuation forces are high and will tend to overcome friction forces

TABLE C.5-2 (Cont'd)

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
SCRDM/SCRD			Redundancy			Tension rod moves relative to sensing tube which moves relative to driveshaft.
		d) Failed bellows	Limit stops and guides	Failed bellows, SCRS Prototype	Post test inspection Structural and fatigue analysis	Components bellows test will identify cycle life
		e) Manufacturing errors		SCRS Prototype	Post test inspection	Testing of SCRS units will identify potential manufacturing errors.
		f) Misalignment	Rotating plug locks		Misalignment analysis	Rotating plug locks preventing accidental plug rotation Misalignment test will identify safety margin and complement analysis.
SCRD	Excessive retarding forces slow tension rod drop	a) Excessive friction from wear, galling	High actuation forces	Latch Scram, SCRS Prototype	SCRS structural analysis Post Test inspection	Component and system tests will determine amount of and effect of wear
		b) Deformation of driveline (thermal, vibration)	High strength materials and heavy sections		SCRS structural analysis	
		c) Seismic	Seismic support		Seismic analysis	
		d) Misalignment	Rotational and axial guides		SCRS structural analysis	

TABLE C.5-2 (Cont'd)

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
SCRD			Guide tube and rod flexibility	Latch, SCRS Prototype	Post test inspection	
			UIS lateral key restraints			
		e) Manufacturing errors		Manufacturing	Post test inspection	Testing of components and SCRS units will identify potential manufacturing errors.
	Excessive friction slows latch release.	a) Self-welding	1718 material cam surfaces			
			Slight pivot of gripper pads break potential welds	Latch	Latch design and test report	Results of component testing show no indication of self-welding.
				SCRS Prototype		Latch and SCRS units will identify effect of self-welding if it occurs.
		b) Misalignment	Heavy cross-section driveline at latch area		SCRS structural analysis	
		c) Particulate deposition	Plant sodium cleaning system	Latch and SCRS Prototype	Post test inspection	Latch and SCRS units testing in prototypic liquid sodium will identify effect of potential particulate deposition
SCA	Duct or guide tube deformation slows insertion.	a) Irradiation induced bowing	Clearance between guide tube and control rod	Guide Tube Bowing	Design Analyses	Testing will support analysis
		b) Seismic	Heavy duct section at load pads	Guide Tube Bowing	SCRS seismic analysis	

TABLE C.5-2 (Cont'd)

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
SCA		c) Swelling and bowing of pins	Low flux at lower pin area when in parked position Stiff bundle tube		Design analyses	
	Excessive retarding forces slow insertion	a) Excessive friction from wear, galling	Hardened wear pads Hydraulic assist force Clearance	SCRS Prototype	Post test inspection Design analysis	Test will determine wear effects and operating margins
		b) Particulate deposition	Plant sodium cleaning system	SCRS Prototype	Post test inspection	Testing in prototypic sodium will identify effects of potential particulate deposition
		c) Seismic	Hydraulic assist force, adequate clearance		SCRS Seismic analysis	
		d) Irradiation swelling of control rod	Parked position raised		Scram clearance analysis	
		e) Manufacturing errors		SCRS Prototype	Manufacturing Pre and post test inspection	Testing of units will identify potential manufacturing errors.
	Loss of hydraulic assist slows insertion	a) Flow blockage or maldistribution	Geometry or flow path opening minimizes blockages Gravity drop capability		Exit flow blockage analysis	Component and SCRS flow tests will support analysis.

TABLE C.5-2 (Cont'd)

Component	Scram Failure Mechanism	General Causative Factor	Design Feature	Test Verification	Analytical Verification	Comments
SCA		b) Weld failure	Stress relievers		Design analysis	Quality control during welding to prevent poor welds.
			Material selection			
		c) Manufacturing errors		SCRS Prototype	Manufacturing	Testing of SCRS units will identify potential manufacturing errors.

TABLE C.5-3

ELECTRICAL SUBSYSTEM MODULE FUNCTIONAL & PERFORMANCE TESTS

Component	Functional Tests	Performance Tests
Trip Comparator	o Trip/Reset Sequence o On-Line Test Sequence* o Bypass Sequence o Manual Trip Function o Setpoint Adjustment	o Trip/Reset Accuracy o Propagation Delay*
Bypass Comparator	o Bypass Permissive Sequence	o Bypass Instatement/ o Removal Accuracy
Buffer	o Signal Transmission	o Accuracy o Isolation
Calculation Units	o Signal Transmission o Potentiometer Adjustability	o Accuracy o Propagation Delay
Logic	o Logic Function	o Propagation Delay
Breaker	o Trip/Reset Function*	o Propagation Delay*
Flux Drawers	o Signal Transmission	o Accuracy o Propagation Delay

*These tests are for the primary subsystem only. All other tests are for both primary and secondary subsystems.

TABLE C.5-4

MECHANICAL SUBSYSTEM DESIGN DIVERISTY

	PCRS	SCRS
Control Assembly (CA)		
Absorber Pin	37	31
Control Rod Geometry[1]	Hexagonal	Circular
51 Number of CA[2]	9	6
Special Feature[3]	Rotational joint in control rod shaft	Latch location at top of CA
Control Rod Driveline (CRD)		
Coupling to control rod	Rigid coupling - released only during refueling	Flexible collet - rod is released at this point for scram and refueling by internal CRDM action
Connection to CRDM	CRD leadscrew to CRDM collapsible rotor roller nuts	Permanent connection to CRDM carriage which traverses only during start up and shutdown
Disconnect from control rod for refueling[4]	Manually - requires special tool	Automatic - same as scram with CRDM deactivation of collet.
Special Features[5]		Heavy CRD wall in the Upper Internal Structure and CA parting plane.

[1]As a result of the difference in control rod geometry, absorber loading and enrichment requirements and effects of transients, the control rod and absorber pin designs in the two systems are completely different.

[2]The larger number of PCA's generally provide for greater redundancy in shutdown capabilities.

[3]The PCA rotational joint eliminates CRD and control rod rotational binding.

[4]The SCRD automatic disconnect feature greatly reduces the per mechanism time for preparation for refueling

[5]The SCRD heavy wall increases the margin against scram latch tension rod drag due to gross Upper Internal Structure to SCA misalignment.

C.6.0 Shutdown Heat Removal System Evaluation

The reliability activity associated with the Shutdown Heat Removal System (SHRS) is the identification of critical failure modes which includes common cause failure modes. Additionally, through feedback of reliability information and data to engineering, design changes to improve reliability can be made. Reliability analysis provides an assessment of the adequacy of the SHRS design to perform its intended functions of decay and sensible heat removal, according to established requirements. Confirmation of design adequacy will be achieved by means of development, acceptance and qualification testing, of selected key items.

To assure timely feedback of data from the test program, the schedule for test activities has been coupled to that for the plant component design, fabrication, installation and operation activities. The testing schedule is such that positive response is possible for the elimination from the plant equipment of any unacceptable features uncovered in the test program. The design and procedural utilization of data from each of the tests is identified at the conclusion of each of the test activity description sections.

C.6.1 Primary Heat Transport System (PHTS)

C.6.1.1 Analysis

Reliability evaluations are being performed on selected SHRS failure modes and components. Two significant evaluations are presented in References 3 and 4.

Reference 3 investigates the probability of loss of the total shutdown heat removal system capability. The failure criterion was assumed to be the bulk in-vessel sodium temperature exceeding 1250°F. This sodium temperature is too low to be associated with in-core sodium boiling and is associated primarily with the ability of reactor and piping structures to retain a pressure boundary and to support the core. Further mitigation is provided by the Direct Heat Removal Service (DHRS) that limits the temperature to 1140°F. Estimates of leakage and rupture of the PHTS and reactor vessel are very low, as documented in Reference 4.

Reference 4 provides an overall assessment of primary piping integrity and focuses on the design, quality assurance, stress analysis and service conditions of the primary piping in terms of the role that each plays in ensuring adequate defense against loss of piping integrity. Supplementing this approach, single point failure reliability analyses were made for the worst locations and loading conditions leading to pipe rupture. Under the assumed presence of a sizeable flaw, calculations of the growth show the critical crack size would not be reached for conservative imposition of loadings. Reference 3 and 4 together provide a total reliability assessment of both system and component features of the SHRS. Both of these documents are updated as the CRBRP Project progresses.

C.6.1.2 Testing

The currently identified testing relating to the PHTS can be divided into two major categories: (a) component performance and acceptance testing and (b) material development tests. Testing in each area has been initiated during the FFTF design phase and is continuing in support of CRBRP. A review of each of these areas is provided below.

Component Testing

Component performance and acceptance PHTS tests on the reactor vessel, primary piping, cold leg check valve, leak detectors, the IHX and the primary pumps are all contributing information in support of SHRS reliability.

A. Reactor Vessel

Component testing supporting the reactor vessel design centers on the outlet nozzles, the sodium makeup nozzle and the upper internals structure (UIS). Component tests of vessel nozzles are in progress as part of the "Validation of High Temperature Design Methods and Criteria" test program. The objectives center on design verification of creep ratchetting due to thermal transients. Strain histories will be recorded at critical nozzle locations. This work will be performed at the Creep Ratchetting Test Facility (CRTF) at ARD. Testing of nozzle attachments is being performed at ORNL in the "High Temperature Structural Design" test program. The emphasis of these tests centers on the inelastic behavior of nozzle attachments.

Numerical studies have been performed to assess the adequacy of the design for the FFTF reactor outlet nozzle. However, the design detail of the CRBRP nozzle liner will be different from FFTF and additional analysis and/or testing will be required to support the CRBRP reliability assessment. Supporting analytic studies are necessary to evaluate the effect of the thermal fluctuation. Material properties needed in this evaluation will be made available in a timely manner from planned test programs.

Prototypic 1/21 scale tests to confirm the thermal adequacy of the location of the DHRS overflow and makeup nozzles has been performed at ARD. These tests have demonstrated that thermal "short circuiting" is about 5 or 6 percent which is considerably less than the 20 percent value to which DHRS is designed. Therefore, the DHRS is conservatively designed.

The 1/21 scale model will also be used in a series of tests to establish the behavior of the outlet plenum flow field in the region of the makeup flow injection. This test is designed to assess the potential of thermal striping initiated failure.

Flow induced vibration has been considered as a potential initiator of failure in the upper internal structure. Outlet plenum flow simulation testing has been performed at HEDL which involves a 1/4 scale

F. Sodium Pumps

The primary and intermediate sodium pumps have major development tests scheduled which will contribute to assuring the reliability of the plant units. A prototype pump sodium test will provide confirmation of design and manufacturing. If unsatisfactory performance is encountered, the data will provide inputs for corrective action to the plant pumps. The corrective actions will be confirmed by water testing the plant units and in-plant sodium testing prior to plant full power operation.

The prototype pump will be subjected to the temperature transients for which the plant units are being designed up to the capability of the test facility. Tests will include endurance runs, thermal transients, speed transients, hydraulic performance, control response and coastdown measurements. These prototype pump sodium tests, curently planned for the time frame late 1981 to mid 1983, may impact the design of the plant units. Design and/or fabrication changes which may be identified by the prototype sodium tests would be retrofitted in final stages of fabrication of plant units. If the water tests indicated problems in the plant units, it would require retrofitting to units in fabrication prior to site delivery.

G. Natural Circulation Verification

The important design feature of natural circulation will be verified through combined computer model development and test verification. Verification of CRBRP's natural circulation capability will be provided by validation of the FORE-2M, DEMO and COBRA-IV computer codes through component test data (pressure drops, pump coastdown tests, decay heat experiments, etc.) and extensive analysis of various aspects such as IHX performance at natural circulation conditions, piping stratification effects, etc. Test and analysis will provide information to verify that natural circulation through the core, primary loops, intermediate loops and steam generators is adequate to remove core heat to the ultimate heat sink. The natural circulation decay heat removal verification plan is presented in Reference 7.

Material Testing

Numerous development programs are also in progress which contribute in a more generic way to reliability assurance of the PHTS. The areas of testing related to reliability include the following:

- Weld joints
- Corrosion effects
- Erosion effects

- Thermal fatigue

- Creep rupture and fatigue

Descriptions of the test programs and the data relative to reliability are provided in the following discussions.

H. Weld Joints

In the area of weld reliability, tests are in progress at ORNL to both develop weld procedures for transition joint welding and study the effectiveness of nondestructive testing on transition welds. Weld reliability is being investigated at ORNL as part of the "CRBRP Transition Joint Welding Program." Cr-Mo steel to stainless steel, Cr-Mo steel to alloy 800H and alloy 800H to stainless steel welds are to be investigated in this study.

Testing of weldments to be used in the design of the reactor vessel thermal liner will be performed at ORNL and ANL. 16-8-2 weld material will be tested extensively to provide data on hardness, tensile properties, creep-rupture properties, creep-fatigue properties, fatigue and metallographic composition. These data will be used to validate the use of 16-8-2 weldments.

Three additional areas of welding will be closely examined for their effect on structural integrity.

1) Material behavior including material properties in the heat affected zone

2) Non-uniformities in geometry including "weld shrinkage"

3) Weld condition including porosity, lack of fusion, cracking and sensitization

I. Corrosion Effects

Corrosion of LMFBR material is the subject of three planned test programs. Inconel 718 is being investigated under the "Component Materials Compatibility Program." Also, SS304 and SS316 are being tested as part of the "Characteristics of Corrosion Due to Leakage of Sodium from a Pipe into Air Test Program." Although the PHTS will exist in an inert environment, corrosion rates identified in air tests will provide conservative data for leaks in the inert environment. In addition, stress corrosion cracking in SS304 is to be examined under the "Caustic Corrosion Product Assisted Fatigue Growth Program." Loading frequency and temperature will be varied in sodium environments.

J. Erosion Effects

Erosion was identified as potentially being significant in many reactor vessel structural members. Data exists which suggests that there is no noticeable erosion effect in sodium flow for velocities below 50 ft/sec. It was determined that erosion could pose a potential threat to structural integrity at higher velocities. Consequently, the core flow of CRBRP is designed conservatively so that the maximum velocity at critical locations is less than 25 ft/sec., and the maximum core sodium velocity is 50 ft/sec.

K. Thermal Fatigue

Quantification of thermal fatigue limits is necessary for reliability assessments. Characterization of material properties is being investigated under the program entitled "High Temperature Tests for Time-Dependent Characteristics of Materials in Sodium" at ARD and ANL. The testing is oriented towards assessing the adequacy of the ASME Code criteria and RDT Standards for design of critical CRBRP components in sodium environments. Tubular and plate specimens are to be tested. Specifically, thermal fatigue will be studied. Additional testing in the "High Temperature Structural Design Program" at ORNL involves the investigation of thermal ratchetting in seamless SS316 pipe.

L. Creep Rupture and Fatigue

Currently at ARD, creep tests with basic specimens and large components are being conducted along with testing of pre-exposed specimens to assess the effect of long-term environmental effects. Conservative minimum creep failure times are also used in the ASME Code, Section III - Case 1592 for high temperature design applications. When the phenomenon of fatigue occurs in high temperature environments, a creep-fatigue interaction may reduce fatigue life. The magnitude of the life reduction depends on the hold times under load in the fatigue cycle. The effect of multiaxial stress states on creep-fatigue on SS304 and SS316 is being studied at ORNL in the "High Temperature Design Program." In addition, the effect of long term exposure on creep-fatigue and multiaxial stress states are under investigation at ARD under the "Component Materials Compatibility Program." These planned test programs are adequate to support conservative reliability analysis.

Fatigue has been extensively studied. Current testing at ARD is oriented towards determining long-term effects on fatigue life by testing samples of SS316 pre-exposed to sodium. At ORNL, the "High Temperature Design Program" will provide data from investigations of fatigue at high temperatures. Currently work is being done at ANL to study low cycle fatigue behavior of SS304 and SS316 at high temperatures. Specifically, the effects on fatigue of roughness, sodium environment, aging and annealing are under investigation.

The analysis of stress rupture depends on the adequate quantification of material properties. Two programs at ORNL involve testing to determine tensile stress rupture properties in uniaxial tests. In the "High Temperature Design Program" tests, the material behavior of SS304 and SS316 will be studied. Another program at ORNL "Mechanical Properties for Structural Materials" is specifically tailored to study the material characterization of SS304 and SS316 reference heats. The heat-to-heat variations in mechanical properties is also under study in this program. The programs in progress along with existing data will be adequate for reliability data needs.

The material data development programs described under Items H through L will provide the information to further qualify the materials and weld processes for the reactor vessel, PHTS piping and IHTS piping. These programs are a part of continuing material technology programs that were initiated for the FFTF program. The data are expected to identify additional margin in the structural components which have been designed on the conservative rules established by the early data and in large part embodied in ASME Code Case 1592. The reactor vessel is in fabrication from early 1976 to early 1980. The PHTS and IHTS piping spool pieces are expected to be in fabrication not earlier than 1982. The data from the materials testing programs have been scheduled to support these schedules.

C.6.2 Intermediate Heat Transport System (IHTS)

C.6.2.1 Analysis

The purpose of the IHTS reliability evaluation is to identify those features of the IHTS which have the maximum impact on system reliability and thereby to permit design action to enhance the reliability of the IHTS piping, the intermediate sodium pump, the expansion tank and the drain valves.

The FMEA for the IHTS is presented in Reference 3. Those failures of the IHTS which result in failure to remove decay heat were analyzed. Results of the FMEA indicate that there are five failure modes which can prevent or adversely affect SHRS operation. These failure modes are: (1) external leakage of sodium piping, (2) significant tube leakage in the IHX, (3) external leakage of the intermediate sodium pump, (4) external leakage of I&C penetrations and (5) external leakage of sodium drain valves. Each of these has been evaluated and does not significantly affect the SHRS function of the IHTS.

The CCFA for the IHTS and its interfaces is in preparation. The significant failure modes which have been identified are: (1) inadvertent operator action or false signals to the actuators of the dump valves, (2) loading from a seismic event and (3) extreme pressure transients.

Postulated dumping of the sodium in all three IHTS loops due to operator error or false actuator action requires the assumption of multiple failures of equipment or multiple operational errors. At least two operations on separate equipment are required to dump a single loop. Postulating this simultaneously for all three loops is not credible. It should be noted that the DHRS will adequately remove the decay heat for postulated events involving the draining of one or more loops.

The IHTS has been conservatively designed to withstand the effects of a conservatively specified earthquake. Therefore, only seismic events substantially greater than the SSE could be postulated to potentially cause failure of all three IHTS loops. Therefore, the potential for common cause failure due to loadings from seismic events is sufficiently remote. Testing of specific components is planned to verify the capability to withstand the SSE imposed loadings (see Section C.6.3.2D).

An extreme IHTS pressure transient could cause failure of the SWRPRS rupture discs.

Numerical Assessment

The quantitative assessment of the probability of failure of the IHTS was determined by inserting predicted equipment failure rates and operational parameters into a mathematical model of the system. The predicted failure rates and corresponding evaluation are presented in Reference 3. The failure probability of the IHTS equipment is low because the mission to remove reactor heat following shutdown requires only the natural circulation capability of the PHTS or the IHTS.

C.6.2.2 Testing

The currently scheduled IHTS testing included main circulation pump testing (discussed in Section C.6.1.2), transition joint development testing and mixing-tee testing.

A. Transition Joint Weld Assembly Tests

The transition joint development is directed towards obtaining information which will provide high confidence in the transition weld region of the IHTS piping. This transition weld is placed in the piping to reduce the differences in thermal expansion between materials to be joined by welding. The 2 1/4 Cr-1Mo material of the steam generator nozzle is joined to alloy 800H which is then welded to the SS316 of the IHTS piping. There are two groups of tests planned for these joints. A group of joints will be exposed to temperature and mechanical load transients more severe than those for which the plant is being designed. The tests will be run to failure and the dominant mode of failure identified. The tests will be accelerated so that failure occurs in about one year rather than the design life. In addition to these complete transition weld assembly tests, there are tests being conducted to establish the proper weld design. This work will contribute to producing weld designs that are appropriate for the transition assembly tests. The transition joint weld design will be completed in late 1978. Fabrication of all spool pieces will be completed in late 1980.

As a final proof test of the transition weld design, prototypic transition joints will be fabricated and used in the prototype steam generator test program. These joints will be exposed to the accelerated testing planned for the steam generator modules and will provide the desirable final confirmation of the transition joint assembly design adequacy.

B. Mixing-Tee Tests

The mixing tee tests were conducted to assure adequate mixing of the two sodium stream flows from each evaporator prior to returning to the intermediate system pump. The temperature difference in the two flows is normally less than 20°F, but can be large if the heat transfer process in one evaporator is interrupted from one of several malfunctions. These tests were conducted to develop a tee which will accommodate these large temperature differences without incurring thermal fatigue failures, should a standard tee design prove to be inadequate.

The initial tests were being run using an 8 inch diameter scale model of the perforated plate mixing tee currently being considered for use in the CRBRP. These tests were conducted with hot water simulating sodium. Tests conducted in water have been demonstrated to be a valid indicator of the mixing characteristics of sodium. The information gathered consisted of temperature fluctuation (amplitude, spatial distribution and frequency), pressure drop across the tee and perforated plate vibration.

The mixing tee development testing is completed. This information, coupled with other sodium mixing tests which have been reported in the literature, will be used to establish the final design.

C.6.3 Steam Generator System (SGS)

C.6.3.1 Analysis

Analysis of the SGS will consist of (1) a determination of the relative probability of occurrence of critical failure modes, (2) further refinement and verification of failure rate data, (3) evaluation of the reliability impact of repairs and (4) analysis of SGS failures that incapacitate one or more heat transport loops.

The FMEA presented in Reference 3 indicates that a potential failure mode of the SGS is a water-to-sodium leakage at the tube to tubesheet weld joints. Other failure modes with the potential to adversely affect the function of the SHRS are: (1) external leakage of the steam generator modules, (2) leakage or inadvertent rupture of the rupture discs in the SWRPRS, (3) inadvertent water dump (operator error of false signal of rupture disc burst), (4) leakage or rupture of water dump valves, (5) leakage or rupture of the sodium drain valves, (6) operator action incapacitating SG loop due to a false signal from the hydrogen leak detection system, (7) leakage, rupture, or internal failure of the steam drum, (8) external leakage or rupture of the recirculation pump, (9) external leakage or rupture of the isolation valves, (10) leakage or rupture of the power relief valves, (11) failure to close the safety valves, (12) external leakage or rupture of the steam or water piping and (13) external leakage or rupture of the instrumentation penetrations. These failures could potentially result in loss of one of the three main HTS heat removal paths, but independence of the IHTS and steam/water loops precludes loss of the other two loops. These failure modes will be addressed in the design of the equipment and the operating procedures.

A CCFA identified that one of the most significant common cause failures could be simultaneous rupture of the rupture discs in SWRPRS under seismic load. A failure of this type could result in draining of the sodium inventory from all three main heat transport loops with subsequent inability to remove heat. Reliability and design verification analysis has been conducted which shows that adequate design margin exists between the peak seismic pressure and minimum disc burst pressure.

Other significant common cause failures which could adversely affect SHRS reliability are: (1) inadvertent water dump (false signals to all three loops), (2) inadvertent closure of isolation valves (false signals to all three loops and (3) water-to-sodium leakage in steam generator modules.

The following paragraphs address measures taken to eliminate these potential common cause failure modes.

A. Inadvertent Water Dump From False Signals to all Three Loops

The water dump valves are in series, and opening requires either (1) individual manual operator switching actions of guarded switches or (2) a signal indicating SGS reaction products vent flow. Each steam generator loop has a separate Class IE logic train for automatic operation making common failure of all three loops highly improbable. Many actions in the proper paired relationship must be taken by the operator to initiate this failure mode.

B. Inadvertent Closure of Isolation Valves From False Signals to All Three Loops

Isolation valves fail open for both loss of electrical power and pneumatics. Automatic closure signals result from indication of initiation of SWRPRS. The most probable cause of isolation valve closure, a false SWRPRS signal, affects only one steam generator loop. Simultaneous failure of all three loops initiating circuitry causing closure of isolation valves in all three loops is sufficiently remote.

C. Water-to-Sodium Leakage in Steam Generator Modules Due to Excessive Thermal Cycling

Thermal cycling and severe transients produce creep-fatigue damage accumulation in the sodium-to-water boundary. The steam generator design and analysis accounts for this effect. In the event that the rate of creep-fatigue damage accumulation is greater than predicted, through-the-wall cracks in the sodium/water boundary could be generated. The occurrence of a through-the-wall crack during plant operation will be detectable. The sensitivity of the detection equipment is such that, under most cases, the leak will be detected at a level where the sodium-water reaction will not result in significant pressure surges. In certain locations, a crack could reach the through-the-wall stage such that a pressure surge occurs before the leak is detected, or before shutdown could be completed. The overpressure protection system is designed to protect the sodium-to-air boundary and IHX boundary from excessive loading from the pressure surge.

In most cases, the leakage rate will be small enough that operation for heat removal would be continued unless larger leaks developed during the required normal module shutdown and isolation. Since the initial leakage rate through cracks formed by the thermal fatigue mechanism is very small, and the probability is extremely small that the through-the-wall stage will be reached simultaneously in 2 or more locations, the loops would not fail simultaneously. This interpretation of the nature of steam generator sodium/water boundary failure is consistent with the experience on LMFBR steam generator leaks in Europe.

The CRBRP SGS is designed to mitigate the hazards associated with a sodium-water reaction as described in Section 5.5 of the PSAR.

Numerical Assessments

Quantification of steam generator system reliability has been accomplished utilizing (1) block diagrams that delineate the redundant and sequential relationships of the SGS constituents, (2) a mathematical model and (3) appropriate equipment failure rate estimates. A calculation of the random independent failure probability of the SGS has been made (Reference 3). This quantitative assessment of steam generator system reliability was used to identify key areas of the SGS requiring further attention.

C.6.3.2 Testing

There is in place a substantial development test program addressing the steam generator modules, sodium-water leak detection and protection against effects of sodium-water reaction. This program is a significant source of information for assessing SGS reliability.

A. Steam Generator Module

There are substantial materials properties and weld development programs which support the development of a reliable heat transfer surface for the steam generator module. The following two tests also provide information applicable to the steam generator module reliability.

1) Shell-Side Hydraulic Model Test

The shell-side flow and tube vibration test were completed in June 1976. This test included a full diameter 757 tube bundle in a shell-side water flow test. Flow rates up to 32,000 gpm were used. Local tube cross flow Reynolds numbers were as high as 2×10^5. The test included simulated (air/water) two phase steam/water flow to determine the significance of tube side vibration excitation. Data have been evaluated for application to both prototype and plant unit design. The evaluations indicate adequacy of the design with respect to failure modes resulting from vibration or shell-side flow distribution.

2) "Few-Tube" Model Test

"Few-Tube" test model steam generator tests have resulted in design changes which will assure that tubes will be free to expand and contract during thermal transient without undue rubbing or galling.

3) DNB Corrosion and Heat Transfer Effects Tests

Two test programs have been conducted in support of the CRBRP evaporator design in the areas of heat transfer and corrosion effects. Tests at Argonne National Laboratory (ANL) over the period of January - December 1976 with a 42 ft (12.8 m) long single tube sodium heated test section produced data to characterize Departure from Nucleate Boiling (DNB), DNB-associated thermal oscillations and post-DNB heat transfer. Subsequent tests conducted in the GE DNB Effects Test Loop during 1976 produced heat transfer data complimentary to the ANL tests and demonstrated that operation with DNB in the evaporator would not produce excessive corrosion under worst case CRBRP thermal oscillations and water chemistry conditions. Following the heat transfer tests, an endurance test was performed during the March - November 1976 period in which the test section was exposed to 4181 hours of steaming time of which 2820 hours were with DNB held within a 24-inch (0.61 m) test zone. Post-test examination showed no localized or accelerated corrosion and based on the uniform corrosion found, a long term life of about 30 years would be predicted for the CRBRP evaporators.

B. Steam Generator Leak Detection System

The steam generator leak detection system development program includes development of instrumentation to (1) detect hydrogen in sodium and (2) detect oxygen in sodium. Programs are in place to develop the detection elements. The detection levels and decision logic for use in the system will also be established through these tests. The tests include operation of the detection system on the "Few-Tube" Test, and at the Experimental Breeder Reactor II at Idaho Falls, Idaho.

C. Burst Disc Testing

The steam generator development program is providing the information for assessing the reliability of burst discs. Tests have been conducted by the manufacturer to confirm that the as-built burst disc design will funtion within specified burst pressure tolerances for the large size discs required in CRBRP. Additional tests are being conducted in conjunction with steam generator component tests. Multiple disc assemblies similar in design to the CRBRP, are being installed in the large leak test rig. Performance of the double reverse buckling disc will be measured during these tests. Results are expected to confirm the design.

C.6.4 Steam Generator Auxiliary Heat Removal System (SGAHRS)

C.6.4.1 Analysis

The SGAHRS provides an auxiliary heat sink for the postulated loss of feedwater or loss of main heat sink. The SGAHRS evaluation will assess the reliability of the Auxiliary Feedwater System (AFWS) and the Protected Air Cooled Condensers (PACCs). The reliability of the main feedwater system and the main heat sinks was evaluated to properly assess SGAHRS.

Potential failure modes identified in Reference 3 include (1) failure of the two AFW motor driven pumps to start and take load, (2) failure of the AFW turbine driven pump to start and take load, (3) failure of the PACCs to operate (includes associated piping and valve failures), (4) failure of the AFW isolation valves to cycle open, (5) external leakage or rupture of AFW piping and valves and (6) external leakage or rupture of the PWST. These failure modes are being addressed in the design of SGAHRS equipment and will be fully resolved in the associated Reliability Design Support Document.

Common cause failures have been evaluated regardless of their probability of occurrence. Parameters considered were common processes, common design properties, common location, common handling, human error test and maintenance acts, external events and extreme environments. Common cause failures identified to date include: (1) AFW control valves fail to remain open due to miscalibration of AFW flow transmitters or I&C on the steam drum, (2) insufficient water due to PWST level miscalibration, (3) AFW isolation valves fail closed due to operator action and (4) PACCs fail due to environmental abnormalities. A major factor contributing to the reliability of the AFWS is its similarity to systems incorporated in current LWRs. The technology used in the design of the AFWS for CRBRP is essentially identical with that used in LWR AFWS design.

Evaluations have been made of the ability of the PHTS, the reactor vessel and the core support structure to sustain an in-vessel sodium temperature of 1140°F. In-vessel sodium would be over 900°F for less than 100 hours with DHRS operation. Creep rupture times exceeding 50000 hours for the reactor vessel and core support structure and 5500 hours for the PHTS were calculated for the most highly stressed conditions. These results indicate an adequate lifetime capability for the reactor system components under DHRS operating conditions.

C.6.5.2 Testing

The 1/21 scale outlet plenum model tests described in Section C.6.1.2A were used to initially evaluate DHRS nozzle locations to assure adequate mixing of the DHRS flow with the PHTS flow. Final confirmation of the thermal adequacy of the location of the CRBRP reactor vessel sodium make-up and overflow nozzles for DHRS was obtained by 1/4 scale model flow tests in the IRFM at HEDL. The DHRS operation was simulated with flows, temperatures and coolant conductivity measured at specific points in the model. Coolant and component surface temperatures were measured in the outlet plenum region and combined with loop coolant temperatures and flow measurement data to provide an analysis base. This experiment confirmed the adequacy of upper plenum mixing for DHRS. Testing is also being conducted on active pumps and valves.

C.6.6 Interfacing Systems

C.6.6.1 Analysis

Assessments are being conducted that provide treatment of all potential failure modes of SHRS interfacing systems or equipment. Preliminary numerical analyses have focused on critical systems and interfaces. Updating of these analyses is a continuing effort.

The SHRS depends upon groups of normal plant operating equipment to provide shutdown heat removal. Important interfaces are described below:

A. Plant Dependence on Electrical Power

For normal plant operation at power, the main feedwater pumps, condensate pumps, condenser circulating water, condenser vacuum pumps, steam generator circulating pumps, primary sodium pumps and intermediate sodium pumps require AC power which is provided by th normal and reserve plant electrical power systems. All sodium primary and intermediate pump pony motors, protected air cooled condenser fans, the motor driven auxiliary feed pumps, the DHRS EM pump and DHRS ABHX fans are provided with both normal and emergency electrical power. All essential instrumentation and control is powered from non-interruptable power supplies."

B. Instrumentation and Control

The reliability of the control and instrumentation involving air supplies, electrical devices, sensing equipment, etc. will be assessed. The DC and AC power subsystems are being analyzed and combined with the instrumentation and control reliability analysis.

C. Balance of Plant (BOP) Systems

One of the key interfaces is the main Condensate and Feedwater System. This system has been subjected to preliminary assessments used in current activities. An assessment to determine the reliability of this system in its role as a heat sink during SHRS operation is being conducted. The Turbine Bypass valving, the condenser and related BOP components are integrated in reliability assessment activities.

SHRS equipment is designed for reliable operation over the full range of operating environments expected within the equipment cells during normal operation. Environmental conditions beyond those specified for the cells could cause degradation of the BOP control systems. The environmental conditions of concern are temperature and chemical contaminents. The effects of these factors on BOP control system reliability is being evaluated. Design, installation and operational features and procedures for these systems will reflect the findings of the evaluations.

C.6.6.2 Testing

Testing identified for interfacing systems is comparable to that associated with current LWR practices augmented by consideration of the potential sodium contamination.

C.6.7 Common Cause Failure Analysis

Results of the preliminary system level study of common cause failures are summarized below.

A. Calibration

All calibration actions are controlled by maintenance and installation procedures. These procedures give detailed information for the calibration and verification of the checkout of each piece of equipment. In addition to accurate and periodic calibration of test equipment and test meters to national standards, calibration of critical sensors to manufacturer's specifications is periodically performed.

Examples of safeguards against inadvertent miscalibration included in the current design are (1) critical instrumentation racks will be locked with one set of keys under administrative control, (2) all valves left in the test position rather than run position after checkout will trigger a warning signal in the main control room and (3) stem lock needle valves that free wheel until a set screw is tightened will be used.

Procedures will be verified during manufacturer's checkout testing, and many of the procedures will receive trial usage in the system level reliability tests. Therefore, a miscalibration could only result from a series of systematic errors caused by the persons involved in the calibration function. The use of these controls makes the likelihood of miscalibration improbable.

B. Environmental Conditions Within Control Room(s)

Included in this category are such factors as temperature and humidity in the area as well as other factors that might be responsible for some local abnormal conditions, such as proximity to heated pipes, magnetic disturbances, etc. Items sensitive to these conditions include: (1) power supplies, (2) switchgears, (3) relays and (4) meters.

A worst-case environment is assumed in these areas and the equipment located there is qualified to operate correctly over the range of environmental conditions specified. Accordingly, as long as the area environment remains within the worst-case limits, the equipment should not be expected to fail for environmental reasons.

C. Failure of Common Air Supplies

The probability of an air supply failure can be made negligible by proper location of equipment and air piping runs. Backup air bottles are provided on safety-related systems (the turbine bypass valves are not safety-related) that must cycle during cooldown. Safety-related valves include accumulators and check valves to protect against an air leak upstream of the check valve(s). Up to ten cycles of operation are available from these backup air bottles. Additionally, to assure that proper installation has been obtained and joints are secure, over-pressure leak checks are planned.

D. Vibration Induced Effects

The components sensitive to vibration will undergo vibrational tests during development testing. Identification of undesirable response characteristics during testing will result in equipment modification so that equipment (e.g., electric equipment cabinets) delivered to the plant will not be inherently vulnerable to vibration.

E. Electrical Power Supply

Certain components in the SHRS require electrical power for proper operation. Power is supplied by the preferred and redundant reserve power lines, the redundant diesel generators and battery supported buses for certain equipment.

The diesel generators and battery supported buses are designed for or protected from the effects that could cause simultaneous failure of both preferred and redundant reserve power lines (seismic, tornado, grid blackout and out of tolerance power). The design of the diesel generators and support equipment to the requirements of IEEE-308 ensures independence of these redundant equipment. Therefore, the potential for common cause failure of necessary electrical power has been made sufficiently remote by the design features included.

Due to the critical nature of supplying power, continuing reliability emphasis will be placed on ensuring the sufficient remoteness of common cause failure as the detailed layout of components and routing of the wiring develops.

C.7.0 Program Evaluation

C.7.1 Reactor Shutdown System

C.7.1.1 Primary Control Rod System

This section summarizes the principal conclusions from analyses and tests performed in support of PCRS reliability assessment.

Analysis

Reliability assessments of the PCRS and interfacing components have verified the reliability adequacy of the system. These analyses have identified those components, features and phenomena upon which the PCRS reliability most relies and/or within which significant uncertainties exist. Supporting analyses and tests have been initiated to resolve uncertainties and establish design margins to confirm PCRS reliability. Design changes have been incorporated to enhance reliability by preventing the occurrence of certain failures or precluding the failures from having a significant impact on scram insertion. PCRS design improvements and interfacing component design features for reliability enhancement are given in Tables C.7-1 and C.7-2, respectively.

Numerical analyses performed for the PCRS have led to the following conclusions:

1) Misalignments within the design envelope resulted in low normal forces retarding scram insertion and should not significantly impact scram reliability.

2) Sufficient margins exist on normal scram sliding friction coefficients such that uncertainties on these data have a negligible effect on scram insertion. Existing LMFBR Base Technology programs will provide acceptable data for CRBRP reliability confirmation.

3) Analyses of CRBRP PCRDM unlatching show acceptable design parameters to achieve unlatching time requirements.

4) Based on analyses and applicable FFTF test data, random independent failure modes for the PCRDM have insignificant impact on scram reliability. Emphasis will be placed on resolving potential common cause failures.

Testing

Conclusions from these tests are summarized below:

A. Duct Crushing Test

The Duct Crushing Test was performed to provide data to support analyses for crushing of core assembly ducts under seismic loading conditions. This test, completed at HEDL, involved transverse loading of EBR-II irradiated ducts.

No brittle fracture was observed in any of the tests. Two bending specimens were loaded to failure. The crush tests (transverse loading) indicated duct deflection capability exceeded deflections predicted for CRBRP seismic conditions. Comparison of available tensile test data for 20 percent CW SS316 and SS304 indicates that SS316 ducts have greater deformation capability than SS304 ducts. The test conditions and loading in this test were based on radial blanket environments. Control assembly ducts have much less severe loading conditions and environments. The tests established that load pad brittle fracture is not a limiting factor in control rod system performance. The ducts can deform to a point where all control rod to duct clearances have been eliminated with no evidence of brittle fracture.

B. Duct Impact Test

The Duct Impact Test to verify the capability of the PCA ducts to accommodate scram arrest impact loads has been completed at HEDL.

In these tests, irradiated EBR-II ducts were impacted by known weights dropped from a range of heights to simulate prototypic and overload conditions. Interim and final examination of both EBR-II ducts tests revealed no failure initiation or cracking of these components. Strain measurements on the two ducts were consistent with each other and increased with increasing impact load. This observation supports the repeatability of the findings. Scanning electron fractography performed on high fluence tensile specimen fracture surfaces at equivalent duct impact strain rates and test temperatures revealed that transgranular channel fracture dominates over all impact conditions.

Results from the duct impact tests have demonstrated that the PCA ducts are not susceptible to brittle fracture under prototypic and overload scram arrest impact loading.

C. Rotational Joint Test

The rotational joint feature was introduced into the PCA design to preclude the transmission of torsional loads from the PRDM via the PCRD to the PCA control rod. It was these torsional loads which led to galling of the control rod outer duct during tests of the FFTF control rods. The rotational joint in the CRBRP control assembly is intended to limit torque transmission to a maximum value of 20 in-lb. At this level of torque transmission, the control rod/outer duct interaction loads are too low to produce galling. Test results have shown that the rotational joint feature limits the maximum breakaway torque to 17 in-lb. During normal operation, the torque transmitted by the joint is further reduced to 8 in-lb max. The tests have demonstrated that the rotational joint effectively precludes the potential for galling of the PCA ducts due to torque loading.

D. Duct Bowing Test

The duct bowing test was conducted to assure that adequate design margins exist for worst case duct bowing predictions and to confirm analytical models for predicting the effects of duct bowing.

The results of this test show that drag forces resulting from duct bowing are negligible (25 lbs. or less) prior to the design limit criterion for three point forced contact between the control rod and outer duct. Worst case design bowing predictions are much less than the design limit criterion. Analysis show that geometrical clearance evaluations can accurately (within 0.005 inch clearance) predict the three point forced contact conditions resulting in increased drag forces. Analysis methods conservatively predict the bowing drag forces. Test result show that flow and control rod velocity have negligible effects on frictional drag forces. The forced contact point occurs at or near full rod insertion with clearances increasing significantly with withdrawal such that negligible drag occurs prior to the last six inches of rod insertion. Consequently, large margins exist for bowing induced failure to shutdown the reactor from both predicted clearances to forced contact and that even after forced contact, the control rod would insert sufficiently to shutdown the reactor.

E. Pin Rupture Test

The pin rupture test was performed to determine the magnitude of control rod duct deformation resulting from potential failure of the absorber pins having high internal pressure due to helium release from B-10 neutron captures.

The test results show acceptable (0.030 inch maximum deformation) inner duct deformation for ruptures of pins containing pressures up to 5000 psi. At expected end-of-life pressures of less than 3500 psi, the duct deformations were even smaller. Ruptures of intentionally faulted pins at different locations within the pin bundle showed no indications of pin deformation as a result of the pressure pulses. These test results show that pin ruptures have negligible for causing a scram failure.

F. Pin Compaction Test

The pin bundle compact test was performed to assess bowed absorber pin interaction effects with the inner duct, pin bundle compressibility and pin to pin contact loads in order to aid verification of analytical models.

Pins prebowed to conditions exceeding design predictions were compressed to design pitch at the top and bottom end caps. Resulting bundle compressibility, pin shapes and typical pin loads were measured. The test results show that the pin bundle is sufficiently compressible that pin bowing will cause negligible deformation of the inner duct and that pin loads are acceptable.

G. Dynamic Seismic Friction Test

The dynamic friction test was performed to obtain effective friction coefficients under impact conditions typical of seismic events and to provide data to assist confirmation of analysis methods.

Geometries tested were a cylindrical rod in three bushings and a hexagon in hexagon configuration. Environments tested were air, argon, water and sodium. A shaker mounted to the test vessel provided the vibrational input at multiple acceleration levels. Measurements included rod drop times and impact load time histories at the bushings. Effective friction coefficients were obtained by simple analyses utilizing the measured impact loads and rod drop times. to check methods used for PCRS seismic scram analyses, these methods were used to predict the impact loads. Effective friction coefficients were than also obtained by utilizing the calculated impact loads with measured rod drop times. Good agreement was obtained between the friction coefficients obtained from measured and calculated impact loads.

The resulting effective friction coefficients were on the order of 0.5 or lower. Utilization of these friction coefficients for PCRS seismic scram speed analyses has shown that design requirements for seismic scram insertion are satisfied.

H. PCRS Friction Couples Tests

Pin or plate friction measurements were obtained for PCRS material couples to obtain sliding friction coefficients for use in normal scram analyses. Utilization of these friction coefficients in scram analyses has shown that scram speed requirements are satisfied even when maximum (3 level) friction coefficients are used in the analysis.

I. PCRS Prototype Design Tests

The PCRDM Accelerated Unlatching Test was performed using a prototype PCRDM/PCRD with a weight simulation for the control rod. Prototypic PCRDM argon and temperature environments were used with a water filled vessel for the PCRD dashpot function. More than twice the design basis lifetime of scrams and travel were completed by including 1868 rod drops and 35, 451 feet of travel. Conditions on nozzle temperatures, nozzle misalignment, internal CRDM pressure and stator cooling beyond the design basis were tested and showed no detrimental effect on PCRDM performance. All tests met design requirements for unlatch time, position indicator accuracy, dashpot final impact velocity and CRDM seal leak rates. Where over twice the service life had no significant effect on PCRDM performance. Post-test inspections confirmed that the testing produced no component failures, no excessive wear and no unusual or unexpected wear patterns.

Phase I of the PCRS Prototype Design Tests has been completed including 470 scrams and 5962 feet of travel. Prototype PCRS components (PCRDM/PCRD/PCA) were tested in a sodium environment under design basis conditions for temperature, flow and misalignment. Sodium exposure, wear and a 42 day hold period had no effect on PCRS performance. Performance characteristics including scram times, position indicator accuracy, dashpot seal leak rates satisfied design requirements.

Maintenance tests using a prototype Disconnect Actuating Tool (DAT) and plant maintenance tools were conducted using draft plant procedures for the tests. Control rod disconnect, installation, removal and power driveline replacements tests were performed. Procedure improvements were identified for DAT and sodium removal operations. Minor lower PCRDM design changes such as more extensive diameters and small reductions in external shielding diameters were identified to facilitate installation. These features have been included in plant units and one prototype test unit. No maintenance relation problems were identified that would affect PCRS functional or scram performance.

J. PCRS System Level Tests

Phase I for both the Real Time and Failed Bellows Tests have been completed, under prototypic conditions. The Real Time Test included 368 scrams and 6053 feet of travel while the Failed Bellows Test included 676 scrams and 9360 feet of travel. Phase I of the Failed Bellows Test was performed to characterize PCRDM performance prior to intentionally failing the bellows for Phase II of this test. Hold time tests including 35 day and 117 day (30 days at full flow conditions) hold periods under various sodium conditions were performed. For all hold time tests, scram times before and after the hold were not significantly (<0.025 seconds to full insertion) different with these time variations being typical of normal variations. Scram times and normal performance for all tests satisfied design requirements. Prototypic maintenance operations for DAT disconnects, installation and removal were performed and showed no impact on PCRS functional or scram performance.

C.7.1.2 Secondary Control Rod System

This section summarizes the conclusions of the reliability evaluations and tests performed in support of the SCRS design.

Analysis

Reliability assessments of the SCRS and interfacing components are being conducted to verify the reliability adequacy of the system. These analyses have identified those components, features and phenomena upon which the SCRS reliability most relies and/or within which significant uncertainties exist. Supporting analyses and tests have been initiated to resolve uncertainties and establish design margins to confirm SCRS reliability. Design changes have been incorporated to enhance reliability by preventing the occurrence of certain failures or precluding the failures from having a significant impact on scram insertion. SCRS design improvements and design features for reliability enhancement are given in Table C.7-3. The interfacing component failure modes and design features are listed in Table C.7-2.

Numerical analyses have been performed for the SCRS. The following conclusions have been reached:

1) Sufficient margins exist on irradiation creep deformation such that control rod duct and guide tube deformation will not exceed clearances provided and impact scram insertion.

2) Based on analyses, random independent failure modes are assessed to have negligible impact on scram reliability.

The SCRS reliability evaluation will be reviewed and updated as required throughout the SCRS development cycle as additional data is made available from testing and from the continued data search and analysis effo·t.

Testing

SCRS testing to date has been directed toward design verification. Four tests have been completed and two are in progress. Design tests of the damper, coil cord, position indication system and the argon system have been completed. The results from these tests were used to optimize the design of the individual component features. Tests in progress are the Latch Test and Latch Seal Test. Results of the Latch Test to date indicated that the latch as currently designed will perform the safety function.

C.7.1.3 Electrical Subsystem

Reliability assessments of the ES and interfacing components have verified the reliability adequacy of the subsystem. These analyses have identified those components, features and phenomena upon which the ES reliability most relies and/or within which significant uncertainties exist.

Numerical analyses performed for the ES have led to the following conclusions:

1) Sufficient failure rate data exists on electronic components to perform meaningful assessments of the random independent failure rate for each of the primary and secondary electrical subsystems. This data is taken from FFTF prototype, qualification and acceptance testing results as well as system and module FMEAs and a numerical module failure rate prediction (using MIL-HDBK-217B as a data source).

2) The redundancy provided in each ES is adequate to reduce random independent failure probability to an appropriate level.

3) Instrument channel monitoring provides significant reduction in dependence on sensor/electronics failure rates.

4) Components with maximum impact of reliability due to failure rate were identified for design consideration (e.g., upgraded MIL-SPEC specification and piece/part changes).

Common cause failure considerations have resulted in the specification of diverse primary and secondary electrical subsystems. Significant conclusions for the hardware implementation resulting from common cause failure considerations to date are discussed in the subsequent section.

A resulting recommendation from ES evaluation was that an extended operations test should be carried out on primary and secondary electrical subsystems. These tests will include all RSS signal conditioning and logic train subsystems.

System Reliability Enhancement Features

A number of reliability improvements over FFTF have been incorporated into the design. For example, the number of primary logic trains has been increased from 2 to 3 to add the capability for on-line testability without bypass.

Design of the FFTF equipment has undergone extensive qualification testing and environmental cycling tests for extended time periods. A reliability enhancement study of the FFTF system was performed by the equipment vendor. The reliability enhancement study determined that the most significant reliability improvement could be gained by increasing the piece/part quality levels and/or the levels to which the piece/parts are screened. Implementing changes to the CRBRP ES equipment based on this study have raised the inherent reliability of the CRBRP equipment.

Piece/part quality levels are provided by using military quality components in major portions of the ES. The use of MIL-SPEC components provides assurance of consistent quality and control during the component manufacturing process.

Preliminary analysis showed that a loss of the -15 volt input to a comparator would tend to prevent that comparator from tripping when required. Even though this single failure would not prevent a reactor scram since the three redundant comparators are powered by separate power supplies, power supply monitors were added. If -15 volt power is lost, all power to the comparator would be cut off by the power supply monitor. Since a removal of all power to the comparator propagates a trip signal on that channel (a safe failure), the effects of a loss of the -15 volt comparator input has been minimized.

The comparators have been designed to minimize the effects of failures in the setpoint circuit. The most likely failure modes for these components were determined and the comparators designed to trip (fail-safe) upon occurrence of these most probable failures.

The packaging of the modules was also modified to increase inherent reliability. Early designs of trip comparators had approximately 130 handwired connections in each module. This has been reduced to six in the present design. Reliability enhancement is realized by the fact that machined, soldered connections are more reliable than handwired connections. The reduction of wiring also minimizes the potential for human error either in initial wiring or future maintenance.

C.7.2 Shutdown Heat Removal System

This section summarizes the principal conclusions from analyses and tests performed in support of the reliability of the SHRS.

Analysis

The analysis that has been completed to date consists of the system level FMEA, system level assessments and probabilistic and structural assessments in support of definition of structural failure in the primary system. These assessments have contributed to the existing base of design information available on SHRS and have supported the identification of changes to design and development programs to enhance SHRS reliability.

References 3 and 4 provided substantial design and reliability information to support the conclusion that large pipe ruptures in the primary system resulting from crack growth due to material flaws can be made sufficiently improbable to exclude large pipe breaks from being a significant problem. The reliability evaluations identified the importance of the role of the leak detection system and quality assurance in attaining the desirable low probabilities for large pipe breaks.

Reference 3 provides the overall numerical assessment of the SHRS reliability. In the report are both a single point estimate of the probability of SHRS performing its mission and significant sensitivity evaluations. The assessment is that with proper attention to design and development activities adequate SHRS reliability can be achieved. The contribution made by DHRS is identified in the report. This finding was part of the information which supported upgrading OHRS to DHRS.

Table C.7-4 provides a list of features which address critical failure modes which have been identified. Table C.7-5 provides a summary list of interfacing systems failure modes and design features for reliability enhancement.

Testing

Many of the materials testing programs contributing information to support the design of SHRS have provided interim data that have been used in the PHTS, IHTS and steam generator system equipment design. A major output from these programs was the 2 1/4 Cr - 1 Mo constituent equations for application to the design of the steam generators.

The shell-side flow and tube vibration test (hydraulic test model) has been completed. The test demonstrated that no adverse vibration effects or flow conditions are present in the reference steam generator module design. Low amplitude (corresponding to low stress levels) vibration was present, however, it was found to have no effect on the reliability of the steam generator module.

The DNB testing for determination of corrosion effects has been completed. Preliminary findings show no evidence of accelerated corrosion or other anomolies.

The integrity of the Sodium to Water/Steam boundary of the steam generator is essential for reliable SHRS operation. The use of microfocus x-ray for 100% examination of steam generator tube-to-tube sheet welds has provided assurance that the welds are free of defects, which would lead to boundary violation.

Testing of tube-to-tube sheet weld development samples has demonstrated that the welds are of the same or greater strength than the parent materials.

LLTR tests have resulted in assurance that the main pressure relief burst discs will limit over-pressures due to sodium/water reactions and that the dual disc design will provide timely pressure relief while providing loop integrity during normal operation. These tests have also shown that multi-tube leakage is highly improbable, even following a guillotine rupture of an adjacent tube.

TABLE C.7-3

SCRS DESIGN FEATURES FOR RELIABILITY ENHANCEMENT

Failure Mode	Design Features
Scram valves jam or stick	Control valve system redesigned with 3 solenoids operating main valves to allow periodic checkout of each electrical channel without causing scram.
Pneumatic actuator piston sticks due to galling	Dashpot type actuator mechanism design replaced by an all metal bellows type design so that the sliding seals are no longer required.
Bellows leak permitting sodium vapor in SCRDM	Lower driveline bellows between tension rod and sensing tube raised to lower head area region from above latch to lower temperature.
Pneumatic cylinder jams due to broken spring debris	Sensing tube and pneumatic cylinder assist springs eliminated.
Tension rod binds to sensing tube/driveshaft due to galling	Wear resistant guide bushings added to maintain position between sliding parts.
Tension rod binds due to lateral distortion	Increased driveline cross section to resist lateral displacement.
Collet gripper fingers self-weld to coupling head	Inconel 718 used to resist self-welding. Cam surfaces are curved to cause severing of any bonds that have developed.
Control rod duct distortion retards scram	Duct to guide tube clearance design is sufficient to minimize contact.
	"Parked" position of control rod raised to a half inch above the top of the core to reduce temperatures and fluence.

TABLE C.7-4

SHRS DESIGN FEATURES
FOR RELIABILITY ENHANCEMENT

Failure Mode	Design Feature
Primary System Leakage	Guard vessels with elevated piping reduced probability of loss of coolant inventory
Intermediate System Leakage	Three independent intermediate system loops are provided in isolated cells.
Loss of Cooling Water	Three independent steam-water heat exchanger systems are provided. Water supplies are provided to assure supply of water to one or all of the three sodium-to-steam/water heat exchanger systems.
Failure of Redundant Forced Circulation Systems	Natural circulation capability in PHTS, IHTS and steam-water system
Common Cause Failure of Independent IHTS Systems	DHRS provided to maintain core coolable geometry
Failure of Electric Motors from AC Power Loss	Battery power supply for short term forced circulation cooling redundant to natural circulation and steam turbine driven AFWS
Failure of all Feedwater Systems	DHRS provides redundancy. PACC's provide long term air cooling following initial cool down.

TABLE C.7-5

SHRS INTERFACING SYSTEMS DESIGN FEATURES
FOR ENHANCEMENT OF RELIABILITY

Failure Mode	Design Features
Failure of Coolant Boundary Component Supports	Ability of piping to tolerate specified snubber and hanger failure.
Off-site AC Power Failure	Redundant diesel generators and short term battery power supplies.
Instrument Gas Supply Systems Failure	Fail-safe designs with backup bottle supply.
Operator Error	Controlled access to critical operations.
Sodium-Water Reaction System Failure	Intermediate System cells provide isolation for independent IHTS systems.
Sodium Fire in IHTS Cell Which Degrades PACC Operation	PACC airflow shuts off when Na fire residues are sensed at PACC inlet to prevent PACC fouling. Operator can override erroneous shut-off signal.

A separate argon supply system similar to but of lower capacity than that described above is provided for GPL-1 because of the remote location of this facility.

Argon gas from these systems is sampled periodically for impurities such as oxygen, moisture, etc., to verify the purity of the gas being supplied to the individual tests.

The Argon Supply Facility is used to provide cover gas for GPL-2 and SASS sodium supply systems. It is also used to supply purge gas for the articles tested in these test rigs. The separate argon supply system located at GPL-1 is used to supply cover gas to GPL-1 for the Dynamic Seismic Friction Test.

E. Sodium Cleaning Facility

The sodium cleaning facility is a separate on-site all-weather facility providing equipment and processes suitable for safe removal and disposal of residual sodium from various sized sodium loop components or test articles.

The facility is equipped primarily to perform sodium removal by moist argon (steam), alcohol and deionized water rinse processes. Equipment is also available for draining sodium from components and for vacuum or inert gas drying of processed components.

The second floor of the facility is equipped with a 2 ton capacity overhead monorail and hoist system for handling and processing components before and after cleaning and also provides a general equipment storage and clean room area for the facility. The first floor features a 350 gallon liquid capacity stainless steel cleaning vessel (LCV) capable of accommodating components up to 20 feet long and 2 feet in diameter.

The Sodium Cleaning Facility will be used to clean test articles and equipment used in the Dynamic Seismic Friction Test and the PCRS System Level Tests.

F. Hydraulic Facility

The hydraulic facility at ARD is a water test facility capable of flow rates up to 5500 gpm at temperatures from 90 to 190°F and pressures up to 200 psi. The facility is comprised of the MPHL and TMHL (defined below) test loops to supply city or demineralizer water at controlled pressures, temperatures and flow rates to various test assemblies and a DAS (defined below) to accept, condition, record and analyze a variety of analog data signals.

This hydraulic facility and the remainder of the hydraulic facilities and data acquisition system described below are used for the Dynamic Seismic Friction Test, the Pin Rupture Test, the Bowed Duct Test and the Control Assembly Hydraulic Test.

G. Thermal Mixing Hydraulic Loop (TMHL)

TMHL is an open recirculation water loop with a 3000 gallon vertical storage tank, a 2000 gpm, 220 foot head centrifugal pump, 3 inch and 6 inch orifice flow meter sections and associated pipe, valving and instrumentation to provide flow and pressure control for a test section. Flow rate can be controlled and monitored from 26 to 2000 gpm within an accuracy of $\pm$ 1.0 percent of the actual flow rate and the pressure drop can be monitored from 0.27 to 100 psid within an accuracy of + 2.0 percent of the actual pressure difference. The water temperature can be controlled from 90 to 180°F. Pumping power is the source of heat input with the temperature being controlled by varying the flow rate to a secondary water-to-water heat exchanger.

H. Multi-Purpose Hydraulic Loop (MPHL)

The MPHL is an open recirculation water loop with a 1000 gallon vertical storage tank, three 2000 gpm, 220 foot head centrifugal pumps, 6 inch and 12 inch orifice flow meter sections and associated pipe, valving and instrumentation to provide flow and pressure control for a test section. The pumps can be arranged to provide maximum flow capabilities of 5500 gpm at 100 psid pump head or 2000 gpm at 200 psid pump head. The water temperature can be controlled from 90°F to 180°F. Pumping power is the source of heat input with temperature being controlled by varying the flow rate to a secondary water-to-water heat exchanger.

I. Data Acquisition System (DAS)

The DAS is located within a controlled atmosphere enclosure in the hydraulic facility. The system is designed to service various test assemblies to provide excitation, conditioning, amplification recording and analysis of data signals.

The system contains signal conditioning instrumentation for resistance bridge, piezoelectric, eddy current and LVDT types of transducers.

Data recording is performed by an FM magnetic tape and an oscillograph chart. Connectors will be provided on all transducers at the individual test area to facilitate maintenance and calibration.

A transducer patching system is used to permit maximum utilization of the signal conditioning for several test programs and to simplify system calibration. An intermediate patch system will provide quick changeover of amplifiers for scaling various levels of input signals and will provide input to the system for high-level transducer signals.

Both the transducer and intermediate patch panels will incorporate test output and input provisions to facilitate calibration and fault isolation.

A high-level patching system is used for routing conditioned signals to the appropriate recording device. The high-level patch permits routing of data to any channel of the recording instruments.

Recorded FM tape data can also be reproduced on the oscillograph or spectrum analyzer using this patching scheme.

Component Test Facilities

A. Dynamic Seismic Friction Test Facility

The Dynamic Seismic Friction Test will use a facility capable of providing (1) support for the test section, (2) a test fluid and (3) a vibration excitation. The test program is subdivided into three phases of which Phase I and III use water as the test fluid. These phases will be carried out in the Hydraulic Facility with the Thermal Mixing Hydraulic Loop described earlier as the source of controlled water flow. Phase II uses a static pool of liquid sodium as the test fluid which will be provided by GPL-1.

Other facilities needed for this test are:

1) Large reaction mass

2) Reciprocating hydraulic actuator (10,000 pound capacity) for the seismic vibration loads

3) Argon supply system for the Phase II sodium test

4) Pressure for the water test

5) Pressure vessel for the sodium test

6) Mechanical actuator for lifting the rod

Test data will be obtained by numerous instrumentation sensors such as accelerometers, displacement and pressure transducers, and strain gages. Analog signals will be conditioned, recorded and analyzed by the Data Acquisition System.

B. Pin Rupture Test Facility

The Pin Rupture Test will use a facility capable of providing (1) support for the test section, (2) a static pool of water and (3) a controlled helium pressure capable of bursting an absorber pin. The test will be located in the Hydraulic Facility described earlier.

The facilities needed for this test are:

1) Support structure

2) Test vessel

3) Helium supply system

4) Booster pump for pressurization of the pin

This test will be performed in the Multi-Purpose Hydraulic Loop facility.

C. Bowed Duct Test Facility

The Bowed Duct Test will use a facility capable of providing (1) hoisting of and support for the test section, (2) lateral forces for bowing the test section, (3) a mechanism for inserting and withdrawing the inner test section and (4) a water test fluid at controlled pressures and flow rates. The test will be performed in the Hydraulic Facility with the TMHL as the source of the water test fluid.

The test facilities needed for this test are:

1) Support structure

2) Test vessel with flexible end sections

3) Hydraulic actuator with pump, valves and piping

4) Outer duct bowing device

5) Inner duct bowing mechanism

D. PCRS Seismic Test Facility

The PCRS Seismic Tests will use a facility where a prototype PCRS may be mounted in the vertical position under prototypic support conditions. Seven electro-hydraulic shaker will be coupled to the support structure and will provide vibration inputs to the PCRS. The test facility is currently under construction.

E. Control Assembly Hydraulic (Flow) Test Facility

The Hydraulic Flow Test will use a facility capable of providing:

1) Support structure with positioning capability

2) Water loop to provide flow rates required

3) Instrumentation to record control assembly response such as pressure drop, vibration

F. Control Assembly Pin Compaction Test

The test facility will be equipped to provide:

1) Means to pre-bow the absorber pins to specified conditions.

2) Test rig to hold the bottom plate and "blossomed" pin bundle with a mechanism to compact the pin bundle to prototypic conditions.

3) Instrumentation to measure pin performance as specified.

G. Control Assembly Rotational Joint Test

The following facilities are needed:

1) Test vessel for the torque transmission tests in sodium.

2) Test vessel to complete accelerated life test in sodium.

3) Test fixture to perform impact tests.

4) Cleaning and inspection facilities to perform wear inspection.

5) Instrumentation to record specified parameters.

System Level Test Facilities

The system level test facility is composed of the features described in the remainder of this section.

A. Test Structure and Enclosure

A tall test structure supports three test control rod test sections and provides a limited area for maintenance and handling. The enclosure provides a sealed containment around the entire test structure and a ventilating system maintains a slight negative pressure within the enclosure for safety purposes.

B. Test Vessel

The test vessel houses a completely prototypic PCRS. Angular and lateral misalignments of the test articles can be accomplished as initial conditions prior to testing under a variety of sodium flow, level and temperature conditions that simulate in-plant operations. Angular and lateral misalignments are controlled by three separate vessel features.

1) Eccentric Flange Interfaces - The test article vertical centerline was offset 1 inch relative to the vessel vertical centerline. By aligning the driveline and control assembly directly under the CRDM, the test article can be installed in perfect vertical alignment. The clamping configuration permits any angular displacement position providing lateral misalignment capability within a 0 to 2 inch range allowed by eccentricity.

2) Misalignment Devices - Misalignment devices are similar in configuration to the stem and operator of a conventional nuclear grade valve. Positioning accuracy was obtained by using a worm gear drive to operate a ball screw, which in turn will cause the anti-rotated ball to translate to the desired position. The irreversible characteristic of the worm gear automatically locks the position of the ball screw nut.

3) Support/Position Plates - Simple machined guide plates were designed for both supporting and positioning the lower end of the CA and lower end of the lower shroud tube. Access to adjust or replace these plates is obtained by their close proximity to the vessel ledge ring and retained by a washer and deformed pin combination that precludes the use of threaded fasteners in sodium.

The CRD test article assembly is provided with upper and lower shroud tubes, which are prototypic in function and design to the CRBRP plant units' shroud tubes.

The upper shroud tube is mounted on a support ledge in the vessel head adapter and extends downward 213.47 inches where it forms a slip connection socket for the lower shroud tube. The vessel head adapter mounting provides capability for positioning and alignment of the shroud tubes relative to the control assembly. The upper shroud will be fabricated from Type 316 SS pipe. In the CRBRP, this shroud tube will be fabricated from Inconel 718.

The lower shroud tube is mounted in the lower shroud tube guide plate with its lower end fixed at reactor baseline elevation (-342.15 inches). From this elevation, the lower shroud tube extends upward 128.5 inches to its slip fit connection with the upper shroud tube. The lower shroud tube guide plate provides positioning and alignment of the shroud tube assembly to ensure a prototypic clearance envelope around the driveline and to maintain a sodium inlet annulus which, in conjunction with exit ports in the upper shroud tube will ensure prototypic sodium flow through the shroud tubes. The lower shroud tube will be fabricated from Inconel 718.

C. Sodium Supply System and Auxiliary Equipment

The sodium supply for the subsystems test utilizes both the GPL-2 and the SASS described earlier. Argon cover gas supply for this test is

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Question CS220.1

Standard Review Plan Sections 3.3, 3.4, 3.5.3, 3.7, (3.7.1 to 3.7.4) and 3.8 (3.8.1 to 3.8.5) have been revised. Your conformance to the revisions in all of the above mentioned sections of SRP is requested.

Response

1. SRP Section 3.3

   PSAR Section 3.3 is consistent with the revisions in SRP Section 3.3.

2. SRP Section 3.4

   PSAR Section 3.4 is consistent with the revisions in SRP Section 3.4.

3. SRP Section 3.5.3

   PSAR Section 3.5.4 has been updated to be consistent with the revisions in SRP Section 3.5.3. Specifically, PSAR Sections 3.5.4.5 and 3.5.4.6 have been updated to be consistent with Appendix A to SRP Section 3.5.3. Table 3.3-1 to PSAR Section 3.3 will not be updated to be consistent with Table 2 to SRP Section 3.5.3. Missile Spectrum A in the November 24, 1975 version of SRP Section 3.5.1.4 was used for design and is acceptable according to the new SRP Section 3.5.1.4.

4. SRP Sections 3.7.1 to 3.7.4

   PSAR Sections 3.7.1 and 3.7.2 is consistent with SRP Sections 3.7.1 and 3.7.2 except that for soil (or rock)-structure interaction analysis only one method was used. The Seismic subsystem analysis in PSAR Section 3.7.3 is consistent with the revisions in SRP Section 3.7.3. PSAR Section 3.7.4 is consistent with the revisions in SRP Section 3.7.4.

5. SRP Section 3.8.1

   This section is not applicable to CRBRP.

6. SRP Section 3.8.2

   PSAR Section 3.8.2 is consistent with the revisions in SRP Section 3.8.2 with the exception of secondary stresses for Levels A, B and C Service Limits which are being studied as part of the response to Question CS220.25.

7. SRP Section 3.8.3

   PSAR Section 3.8.3 is consistent with the revisions in SRP Section 3.8.3 except for the following PSAR subsections which have been updated to be consistent with the SRP.

   a. PSAR Sections 3.8.3.2 and 3.8.3.3 reference compliance with ACI-349 and RG 1.142 as given in subsections II.2 and II.3 of SRP Section 3.8.3.

b. PSAR Appendix 3.8-D has been added to incorporate the requirements of Masonry Walls.

8. SRP Section 3.8.4

PSAR Section 3.8.4 is consistent with the revisions in SRP Section 3.8.4 except for the following PSAR subsections which have been updated to be consistent with the SRP.

a. PSAR Sections 3.8.4.2 and 3.8.4.3 reference compliance with ACI-349 and RG 1.142 as given in subsections II.2 and II.3 of SRP Section 3.8.4.

b. Appropriate sections of PSAR have been updated to add requirements of Masonry Walls to be consistent with Appendix A to SRP Section 3.8.4.

9. SRP Section 3.8.5

PSAR Section 3.8.5 is consistent with SRP Section 3.8.5.

Question CS220.2 (3.5.4.1 & 3.5.4.2)

The revised Petry equation for penetration depth as a function of velocity seems to have been copied incorrectly in that the term in the exponential is dimensionally incorrect and the term V', which is a logarithm, is given with a dimension (the velocity dimension should be incorporated in K). Further the K in the text does not agree with the K in Figure 3.5-1. It is requested that corrections be made. Indicate how you calculate d_m for a noncylindrical or nonspherical projectile?

Also indicate if the wall thicknesses you determined meet the requirements as shown in Table 1 on Page 3.5.3-6 of the revised SRP Section 3.5.

Response:

PSAR Section 3.5.4.1 will be revised to show that the term V' does not have the dimension ft/sec and that the term a should be equated to Tp/KApV'. The value of K in PSAR Figure 3.5-1 will be corrected to 2.76×10^{-3} to agree with PSAR Section 3.5.4.1.

For a noncylindrical or nonspherical projectile or missile, d_m is calculated by determining the equivalent diameter of a noncircular missile:

$$d_m = \sqrt{\frac{4A}{\pi}}$$

where A = cross-sectional area of missile

The wall thicknesses determined in the CRBRP design are consistent with the requirements shown in Table 1 on page 3.5.3-6 of the revised SRP. For concrete strength of 4000 psi, the minimum wall thickness designed is 27" which is greater than the required minimum thickness of 20". The minimum roof thickness of 27 as designed is more than the required minimum thickness of 16".

Question CS220.3 (3.5.4.5)

On page 3.5-13b ductility ratios for concrete and steel are listed. Some of these ratios are different from those specified in Appendix A to SRP Section 3.5.3. Conformance to SRP Section 3.5.3 ductility ratios is requested unless justification for deviation is provided.

Response:

PSAR page 3.5-13b is revised to incorporate ductility ratios used in the design of concrete and steel structures.

Question CS220.4

Seismic design is presented in Section 3.7 supplemented by Appendix 3.7-A. However, a review of the section and its Appendix will reveal that there is quite some repetition in the Appendix of the materials presented in the body of the section.

The Appendix also mentions loads and load combinations which are delineated in Section 3.8. The presentation of the materials in this manner not ony consumes effort in preparation by you and in review by the staff unnecessarily, but also may lead to contradiction and confusion. It is therefore proposed that Appendix 3.7-A be revised to eliminate materials which are not contained in Section 3.7 and 3.8.

Response:

Appendix 3.7-A is a self-contained appendix to the body of the PSAR Section 3.7. As an appendix, it is, by its very intent, not required to eliminate information presented in the body of the PSAR but to supplement it with more details. This necessitates a certain amount of duplication and repetition. We believe that the additional information will be helpful in the NRC review. To assist the NRC staff in their review, the applicant has prepared a cross reference (ref. Table CS220.4-1) of information for your use.

Table CS220.4-1

	SUBJECT	PSAR SECTION(S)
I.	SEISMIC DESIGN PARAMETERS	2.5, "GEOLOGY AND SEISMOLOGY" 3.7-A PARAGRAPHS 3 AND 4, "SITE DESCRIPTION" AND "OPERATING BASIS FOR SAFE SHUTDOWN EARTHQUAKES".
	1. <u>DESIGN GROUND MOTION</u>	
	a) Design Response Spectra	2.5.2.10 and 2.5.2.11 3.7.1.1 and 3.7.1.2
	b) Design Time History	3.7-A.4.3 3.7.2.1.1, 3.7.2.1.2 and 3.7.2.3
	2. <u>CRITICAL DAMPING VALUES</u>	3.7.1.3
	3. <u>SUPPORTING MEDIA FOR CATEGORY I STRUCTURES</u>	3.7.1.5, 3.7.1.6

SUBJECT	PSAR SECTION(S)
II. SEISMIC SYSTEM ANALYSIS	
1. SEISMIC ANALYSIS METHODS	
a) Dynamic Analysis Methods	3.7.2, 3.7-A.6 and 3.7-A-A
b) Equivalent Static Load Method	3.7.2.1.2 and 3.7.A.2
2. NATURAL FREQUENCIES AND RESPONSE LOADS	3.7.2.2
3. PROCEDURES USED FOR ANALYTICAL MODELING	3.7-A-A
4. SOIL-STRUCTURE INTERACTION	3.7-A-C, 3.7.1.6
5. DEVELOPMENT OF FLOOR RESPONSE SPECTRA	3.7.2.6
6. THREE COMPONENTS OF EARTHQUAKE MOTION	3.7.2.1.1 and 3.7-A-B
7. COMBINATION OF MODEL RESPONSES	3.7-A-A.1.3 3.7.3.7
8. INTERACTION OF NON-CATEGORY I STRUCTURES WITH CATEGORY I STRUCTURES	3.7.3.13 and 3.7-A.6
9. EFFECTS OF PARAMETER VARIATIONS ON FLOOR RESPONSE SPECTRA	3.7.2.1.1 and 3.7.2.8
10. USE OF EQUIVALENT STATIC FACTORS	3.7.3.9 and 3.7-A-A.2
11. METHODS USED TO ACCOUNT FOR TORSIONAL EFFECTS	3.7.3.11
12. COMPARISON OF RESPONSES	3.7.2.11
13. ANALYSIS PROCEDURE FOR DAMPING	3.7.2.14
14. DETERMINATION OF CATEGORY I STRUCTURE OVERTURNING MOMENTS	3.7.2.13

SUBJECT	PSAR SECTION(S)
III. SEISMIC SUBSYSTEM ANALYSIS	
1. SEISMIC ANALYSIS METHOD	3.7.2.1
2. DETERMINATION OF NUMBER EARTHQUAKE CYCLES	3.7.3.1
3. PROCEDURES USED FOR ANALYTICAL MODELING	3.7-A-A
4. BASIS FOR SELECTION OF FREQUENCIES	3.7.3.2
5. ANALYSIS PROCEDURE FOR DAMPING	3.7.2.14
6. THREE COMPONENTS OF EARTHQUAKE MOTION	3.7.2.1.1 and 3.7-A-B
7. COMBINATION OF MODEL RESPONSES	3.7.3.4
8. INTERACTION OF OTHER SYSTEMS WITH CATEGORY I SYSTEMS	3.7.3.13
9. MULTIPLE-SUPPORTED EQUIPMENT AND COMPONENTS WITH DISTINCT INPUTS	3.7.2.14 and 3.7.2.7
10. USE OF EQUIVALENT STATIC FACTORS	3.7.2.1.2 and 3.7-A.2
11. TORSIONAL EFFECTS OF ECCENTRIC MASSES	3.7.3.11
12. CATEGORY I BURIED PIPING, CONDUIT AND TUNNELS	3.7-A-C.3
13. METHODS FOR SEISMIC ANALYSIS OF CATEGORY I DEMO.	3.7.2.12

SUBJECT	PSAR SECTION(S)
IV. SEISMIC INSTRUMENTATION	
1. COMPARISON WITH REG. GUIDE 1.12	3.7.4.1
2. LOCATION AND DESCRIPTION OF INSTRUMENTATION	3.7.4.2
3. CONTROL ROOM OPERATOR NOTIFICATION	3.7.4.3
4. COMPARISON OF MEASURE AND PREDICTED RESPONSES	3.7.4.4
5. INSERVICE SURVEILLANCE	INSERVICE INSPECTION PROGRAM FOR SEISMIC INSTRUMENTATION WILL BE PROVIDED IN SECTION 16.4 OF THE PSAR IN A FUTURE AMENDMENT.

Question CS220.5

The major seismic Category I structures of the CRBR plant are supported on a common basemat founded on competent rock with an embedment of 100 ft of back fill. Under such a condition, it appears most appropriate to consider the structures as fixed at the foundation.

The embedment effect can be accounted for by considering the soil-structure interaction between the lateral earth pressure and the structure in contact. The seismic input motion should be applied at the foundation level. The applicant has considered an analysis in which there is soil (rock) structure interaction at foundation level as well as on the lateral side with the seismic input motion applied at the finished grade level. In staff's opinion such an analysis does not represent the realistic condition and the complexity of the analysis as used by the applicant precludes a prior assessment of the adequacy of the method for staff review. As a resolution of staff's concern it is required that seismic Category I structures, systems and components be designed to seismic effects obtained by enveloping the results of applicant's and the fixed base approach as stated above or equivalent.

Response:

The major Category I structures of CRBRP with the exception of the Diesel Generator Building, are supported on a common basemat founded on rock with an average shear wave velocity of 4000 ft/sec. The material above the elevation of the foundation mat (the embedment material) consists of sound and weathered rock, lean concrete fill and compacted Class A backfill.

The input motions were applied at the foundation level and not at grade level (Section 3.7.1.1 of the PSAR).

The justification for using lumped springs and dashpots in lieu of "fixed" base for rock-structure interaction is given below.

1) In the seismic analysis of the CRBRP Nuclear Island, the actual stiffness of the foundation material was evaluated in terms of equivalent springs and dampers and on this basis the seismic analysis was performed.

2) In the CRBRP seismic analysis it was considered that using a fixed based analysis was unwarranted, since due to the large size and stiffness of the structure that is comprised of all the Nuclear Island buildings, some interaction was expected between the foundation rock and the structure. This was confirmed by the calculated responses at the foundation mat which differed from the "free field" responses (Figure Q220.5-1)

   Additional analysis, using a different analytical approach (the Computer Program FLUSH) showed that the free-field and in-structure seismic responses at the foundation level differed, confirming that rock(soil) structure interaction will occur (Figure Q220.5-2).

3) To calculate the foundation springs, because of the irregular layering of the site, and variation of foundation properties, a static finite element method was used. (Section 3.7.1.6 of the PSAR).

4) Calculations using the theoretical half-space equations were also performed (Response to NRC Question 130.53). The results of the elastic half-space calculations verified, within reasonable limits, the values obtained from the static finite element calculations.

5) To account for uncertainity in foundation material properties three sets of spring stiffnesses were calculated: for upper bound, average and lower bound of material properties. Analyses showed that responses with average foundation material properties were enveloped by those for the upper and lower bound. Two complete analyses were then performed using the upper and lower bound material properties and the responses were enveloped.

6) The radiation damping was calculated based on the half-space equations for equivalent material properties deducted from the spring constants obtained by the finite element calculations. This approach was validated by the fact that the results from the finite element and half-space calculations were in good agreement.

SRP 3.7.2 of NUREG-0800 defines various acceptable methods for modeling and analyzing soil structure interaction effects. As described above, the CRBRP seismic design conforms to acceptable methods that are representative of the site geologic conditions and the Nuclear Island structures. Appropriate conservatism has been included in the model to account for a range of foundation material properties that were developed from a rigorous subsurface investigation. The SRP defines "a fixed base assumption" as an acceptable basis for modeling if the structures are supported on rock. Such an assumption is not considered reasonable for CRBRP since analysis has confirmed that interaction will occur and a more realistic and conservative representation of the rock and embedment conditions has been accounted for in the seismic design. The average shear wave velocity of 4000 fps is not characteristic of a hard rock material that would be consistent with a "fixed" base assumption.

The requirement of a fixed base approach for CRBRP is inappropriate and arbitrary. The analysis of a "fixed" base model, therefore, will not be considered for CRBRP.

- o The largest historical earthquake in the tectonic province was assumed to occur in the CRBRP site.

- o SSE maximum ground acceleration was increased from 0.18g to 0.25g.

- Design Response Spectra consist of wide band envelope spectra based on statistical studies of many past earthquake records.

- Artificial acceleration time-histories used in the seismic analysis envelope and for most frequencies are above the Design Response Spectra.

- Floor response spectra are envelopes of two independent analyses using lower and upper bound of soil-rock properties.

- Floor response Spectra were widened at peaks and smoothed.

An independent finite element analysis using the FLUSH program was performed to compare floor response spectra with the CRBRP design spectra. The CRBRP spectra in essence envelopes the calculated spectra and spectra generated from the CRBRP and finite element analyses are very similar (Figure Q220.5-3).

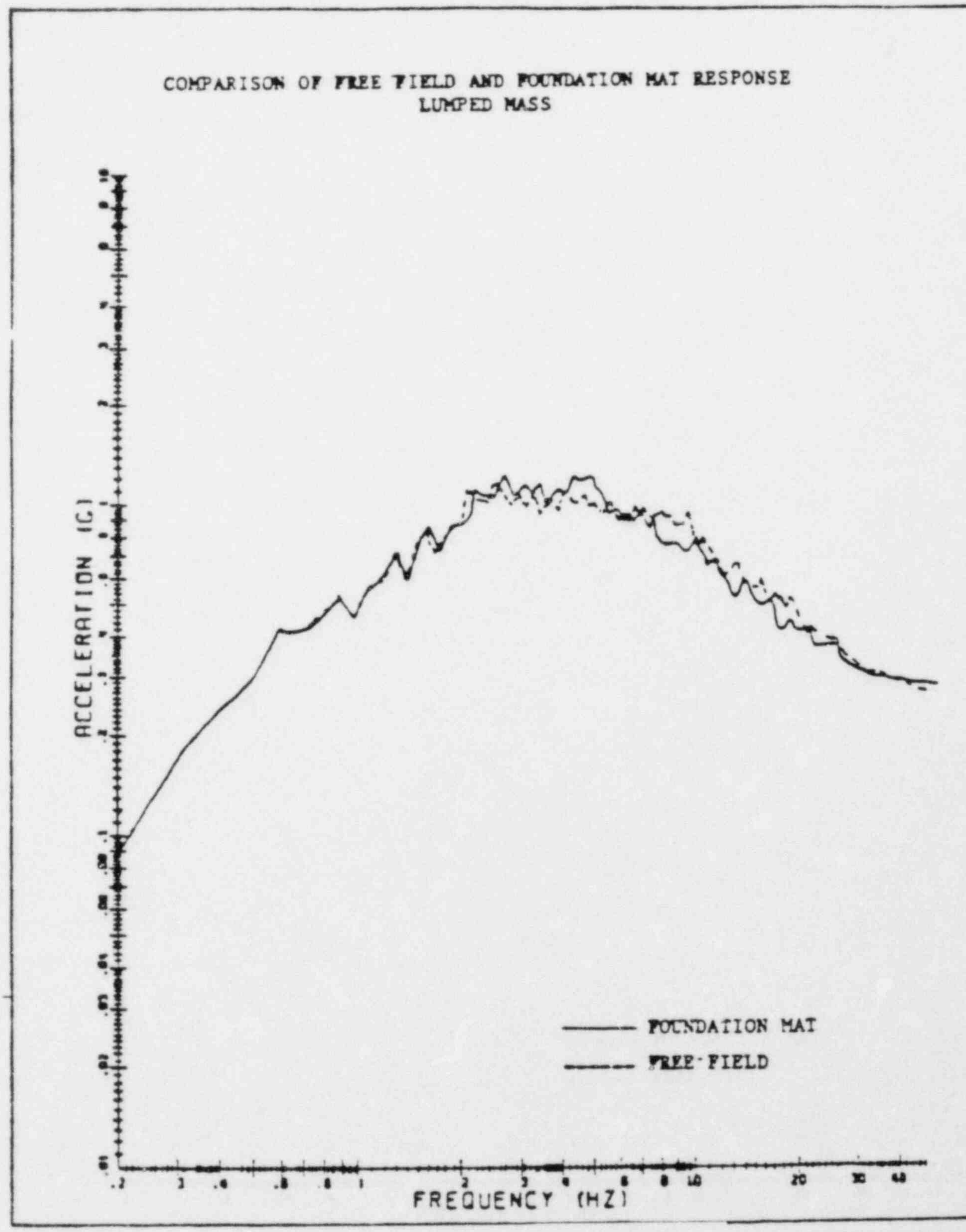


FIGURE QCS220.5-1

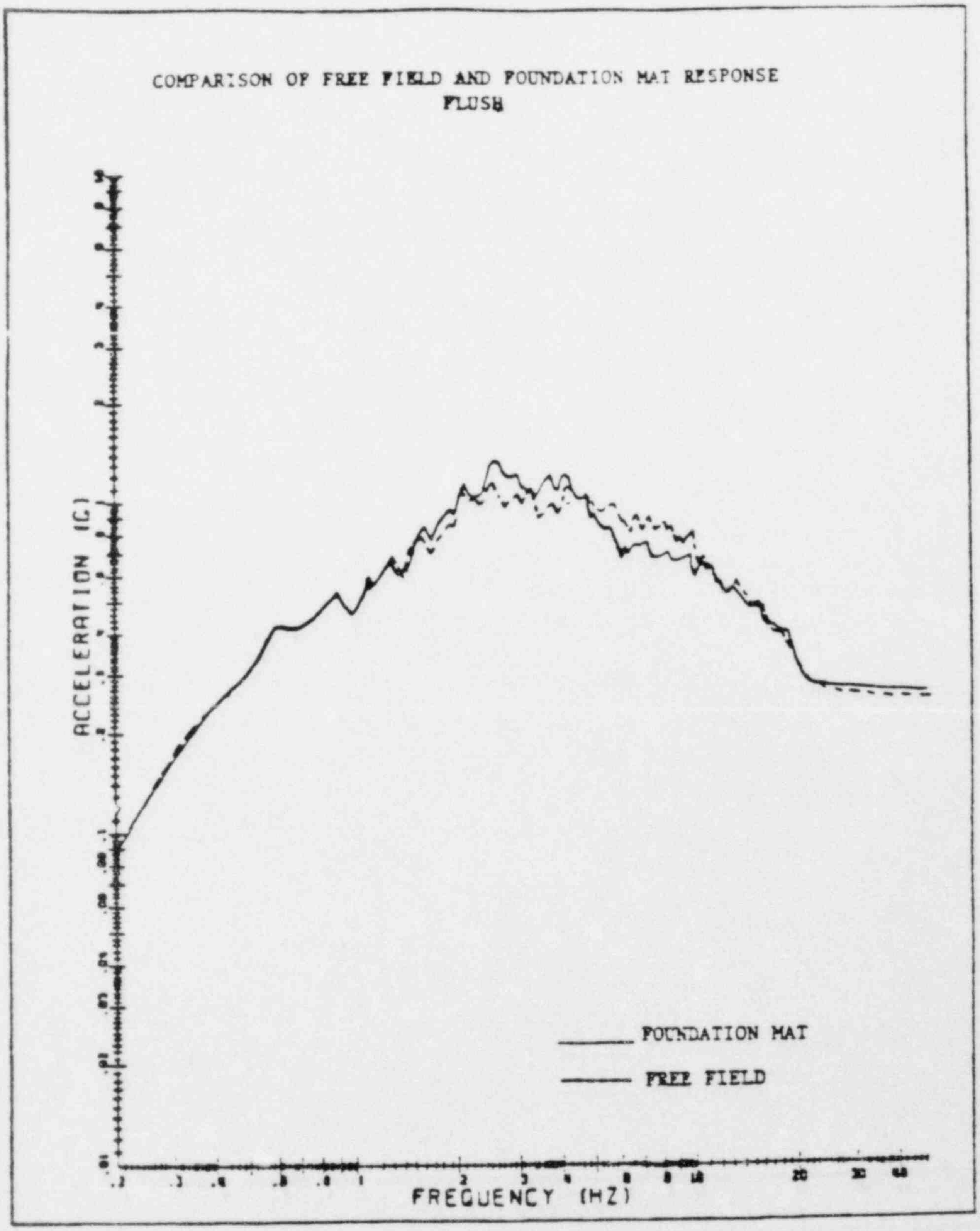


FIGURE QCS220.5-2

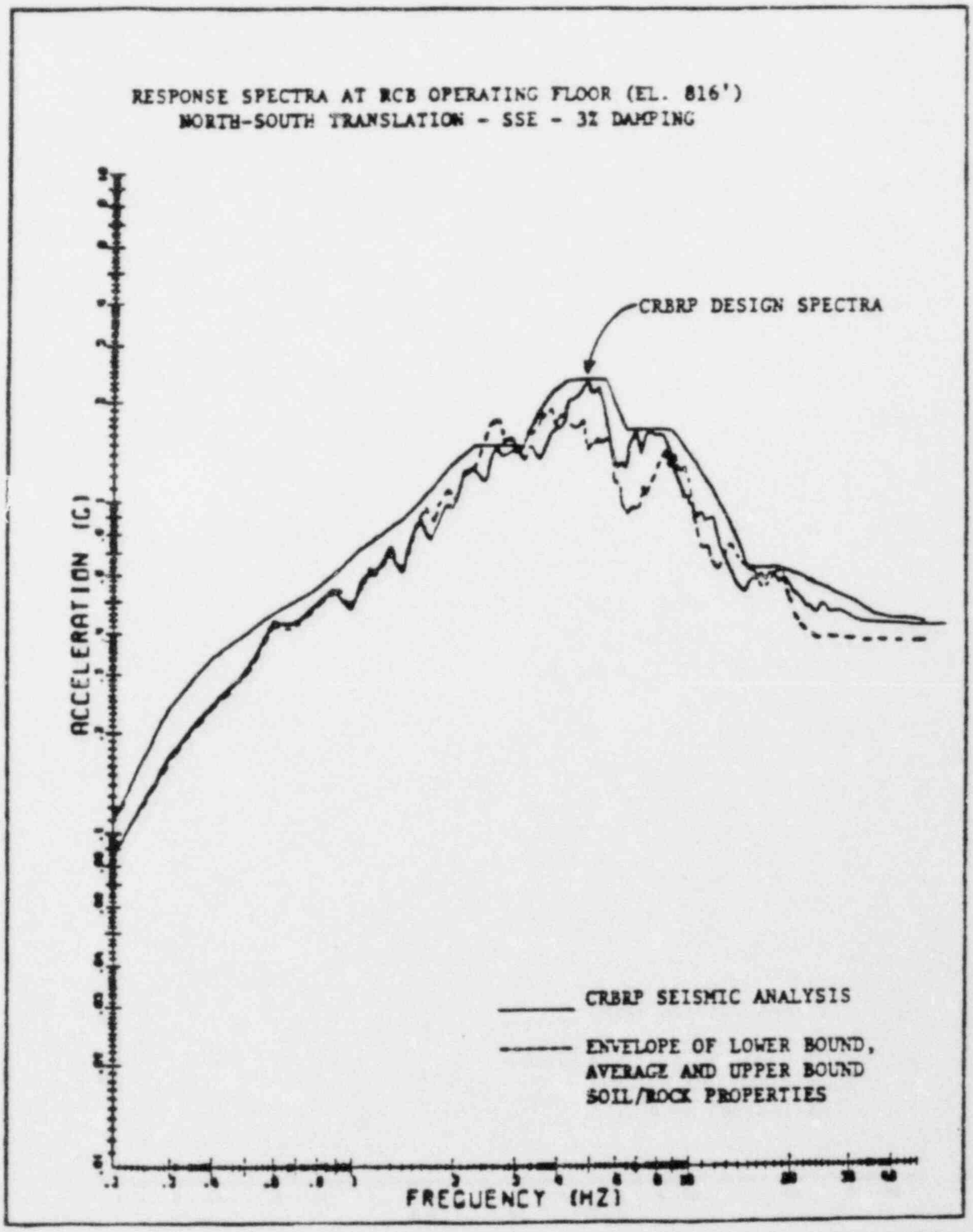


FIGURE QCS220.5-3

Question CS220.6 (3.7.1.1)

It is stated that for a lumped-mass-spring type of models the seismic design response spectra will be applied at the foundation. The mathematical models as shown in Figures 3.7-16, 3.7-16A, and 3.7-16B are the lumped-mass-spring type. Indicate how the springs and dashpots representing soil are derived from a static finite element model. Provide a description in detail.

Further, the mathematical models in Figures 3.7-16, 3.7-16A, and 3.7-16B lack numerical details. No one could judge the adequacy of plans for a plant model based on the material given on these diagrams. A full discussion with tables should be provided delineating the numbers, their meanings, etc.

Response:

A detailed description on how foundation springs and dashpots were calculated is given in Section 3.7.1.6 of the PSAR (Pages 3.7-3a, 3.7-3b, 3.7-3c, and 3.7-4.)

The attached diagrams are the updated mathematical models used in the seismic analysis of the Nuclear Island buildings. Figures 3.7-16, 3.7-16a and 3.7-16b show respectively the mathematical models for the analyses in the North-South, East-West and Vertical directions.

The mathematical models consist of four main parts:

1) The Reactor Service Building

2) The Confinement Structure

3) The Reactor Containment Building

4) The Steam Generator, Electrical Equipment and Control Buildings

The Reactor Vessel and the polar crane are coupled by means of simplified lumped-mass models.

The nodes or mass points correspond to the locations of centers of mass and were selected in general, at the floor elevations. For each of the horizontal analyses (North-South or East-West) three dynamic degrees of freedom per node were allowed (translation, rotation and torsion). For the vertical analysis, one dynamic degree of freedom (translation) was allowed. The beam elements which connect the different nodes vertically are located at the shear centers of their sections and are characterized by areas, shear areas and moments of inertia (for bending and torsion) of the members and by the modulus of elasticity and Poisson's ratios of the material. The ends of beam elements are connected to the mass points by horizontal rigid members. The four parts of the model are supported by the foundation mat which is assumed to be rigid. This assumption is justified because the mat acts as a diaphragm and is stiffened by the vertical walls of the buildings. The buildings above the mat are interconnected by flexible ties which include cross-coupling between the interconnected nodes. The Reactor Containment Building is connected to the other elements only at the mat and operating floor levels. In the analysis for the vertical direction, the steel containment dome was idealized by using equivalent springs which account for the "breathing" of the dome during a vertical vibration. "Breathing" is a shell mode of vibration that the dome experiences under a vertical motion.

Figure 3.7-16c shows a plan of the Nuclear Island and the system of coordinates. Table 3.7-7 gives the coordinates of the mass points (nodes).

PSAR Section 3.7.2.1.1 and the referenced tables and figures have been updated to include the design information discussed above.

Question CS220.7

In Table 3.7-2A damping values are related to the shear strain values. Indicate how such relations are obtained.

Response:

The dynamic properties for compacted granular fill used in seismic des... were established based on laboratory and in situ testing on compacted granular ..ll used at another nuclear plant site. Details were provided in response to Question CS324.7 and were subsequently incorporated into Section 2.5.4.5.1.5 of the PSAR.

PSAR Table 3.7-2A has been revised for consistency with values in Section 2.5.4.5.1.5.

Question CS220.8 (3.7.1.6)

It is stated that the input motions shall be applied at the surface level (finished grade) on an assumed rock outcrop and shall consists of the rock motion used in the analysis of the Nuclear Island and that no credit shall be given for soil cover on overburden in the deconvolution. Clarify this statement and provide a full discussion on how the analysis will be done.

Response:

The application of the input motion at the top of an assumed rock outcrop is the result of NRC instructions (Ref. QCS220.8-1) with regard to a FLUSH analysis of the Nuclear Island contemplated during 1976.

For consistency in the input, this method was used in the analysis of the seismic Category III structures (Turbine Generator and Radwaste Buildings,) adjacent to the Nuclear Island. The computer program FLUSH was used in these analyses.

The deconvolution consists of determining the input motion to be used at the base of the FLUSH mathematical model (of the soil-structure system) for an input motion defined at the surface. This was doen by using a profile representative of the "free-field" in which the rock was extended to the finished grade, disregarding the overburden or backfill. The FLUSH models for the soil-structure interaction, however, represented the actual profile including soil and rock. Strain dependent properties of the soil were used. Analyses were performed for a range of soil-rock properties and the results enveloped.

Since the FLUSH analysis is two-dimensional, separate models for the North-South and East-West directions are required.

For the Diesel Generator Building, which is supported on soil, the soil-structure interaction will be performed with FLUSH. In the deconvolution, the input motion will be applied at the finished grade using the actual profile including soil and rock. The response spectra at the "free-field" foundation level will envelope the Design Response Spectra.

PSAR Sections 3.7.1.6 and 3.7.2.1.1 have been upgraded to include the above information.

Reference:

QCS220.8-1 Letter of July 2, 1976 from Themis P. Spies of USNRC to Lochlin W. Caffey of CRBRP.

Question CS220.9 (3.7.1.6)

On page 3.7-3a, you mentioned the backfill of lean concrete and the use of compact Class A fill for the space between the side of excavation and the plant structure. Discuss the merit of such a fill and how it is considered in your analysis.

Response:

Lean concrete backfill is being used to fill the gap between the vertical excavation and the structures. The lean concrete is used to fill narrow gaps between the rock and structure where the construction of compacted fill will be difficult. For larger areas compacted Class A fill is used. Class A fill is placed around the structures in areas where a sloped excavation is used, i.e. the west side of the Nuclear Island and on all sides above the weathered rock planes. In the calculation of the lumped springs, the fill concrete was assumed to have the properties of the surrounding rock. The mathematical models accounted for the properties of the compacted backfill in the appropriate areas of the profile.

Question CS220.10 (3.7.1.6)

In last paragraph on page 3.7-4, you stated that a fixed base approach would be justified. However, in order to account for soil-structure interaction effects, you made a number of simplifying assumptions and also conducted a scoping study to take into account variations in spring constants and damping values. Indicate if you have taken the fixed base condition into consideration in your scoping study since this is more representative of the actual condition.

Response:

See the response to Question CS220.5.

Question CS220.11(a) (3.7.1.6)

How different are the vertical translation soil-basemat interaction spring constants calculated from the N.S. direction and the E. W direction soil foundation interaction models? What physical effects are implied by this difference and how are these effects accounted for elsewhere?

Response

For the upper bound of foundation material properties, the vertical spring constant calculated from the east-west model was 1.7% higher than that from the north-south model. For the lower bound, the north-south model gave a value 5.5% higher than the east-west model. The physical effects of these differences are small and are due to the fact that the North-South and East-West profiles of the site are different.

Question CS220.11(b) (3.7.1.6)

On top of page 3.7-4, the word "rotation" is probably missing from the last sentence of the first paragraph. Was the unit of the rotation applied to an element that was "rigid" by comparison, or by a rigid link method? or perhaps another method?

Response

The word "rotation" should be inserted between "unit" and "was" on line 8 of the first paragraph on page 3.7-4. The PSAR, page 3.7-4 has been revised to correct this error of omission.

The rotation was applied as tangential displacements at the boundary nodes of the structure, which would correspond to a net rotation applied at the centerline of the equivalent mass.

Question CS220.11(c) (3.7.1.6)

The finite element models in Figures 3.7-21, 3.7-23, and 3.7-24 are labeled "not to scale" yet the element aspect ratios appear reasonable. What are the magnitudes of the average and extreme aspect ratios for these meshes?

Response

The two horizontal models (N-S and E-W, Figures 3.7-21 and 3.7-23) have been drawn to scale, traced from computer generated plots. A revised sketch of the torsion spring model (drawn to scale) is attached.

The average aspect ratio is approximately 1.5:1. The aspect ratio generally does not exceed 1.25:1 within a distance of one width of the opening. The worst aspect ratios are in the vicinity of the outer boundaries and in no case, does the aspect ratio exceed 10 to 1.

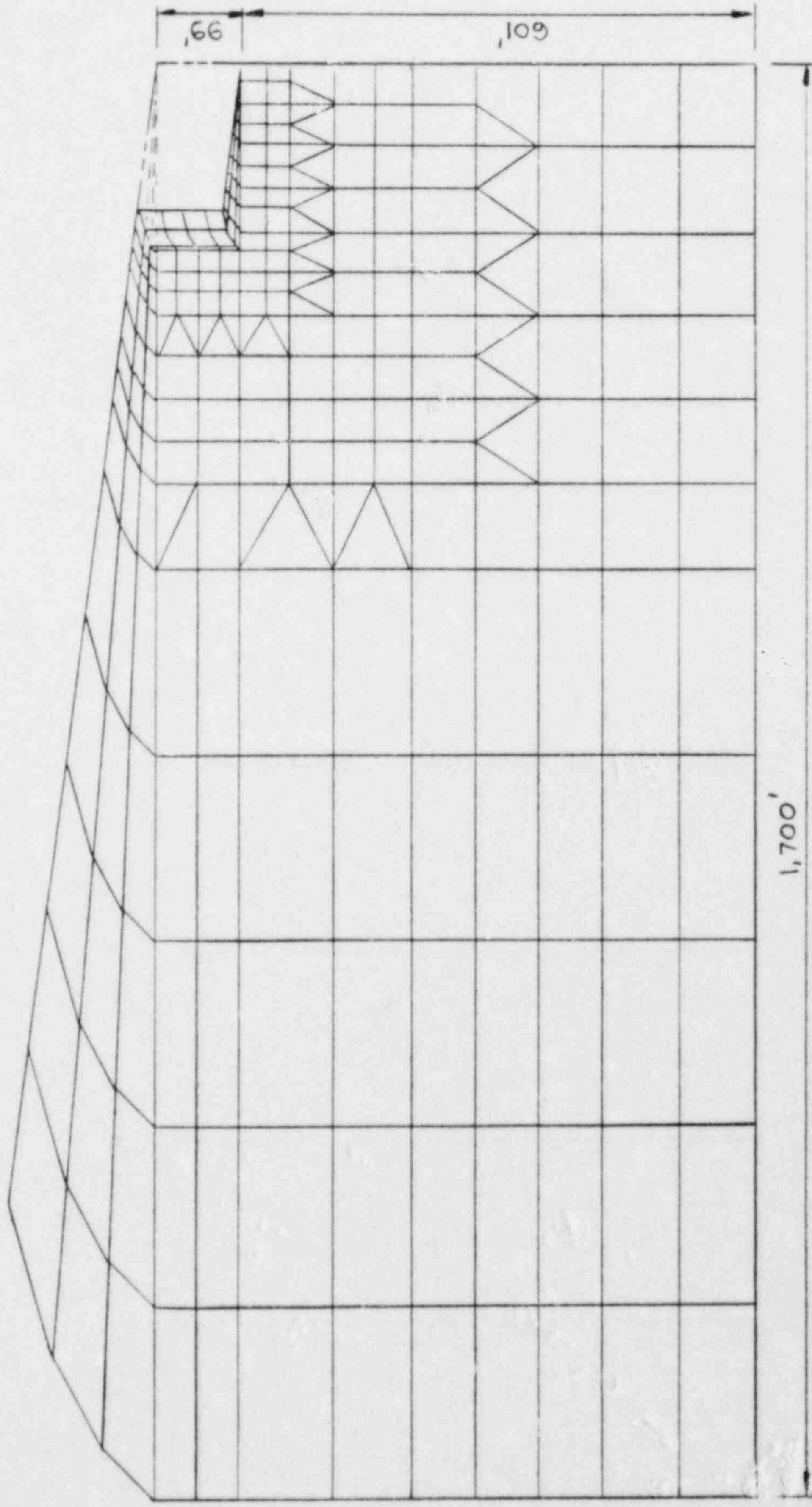


FIGURE 220.11(c)-1 FINITE ELEMENT MODEL - TORSIONAL SPRING

Question CS220.11(d) (3.7.1.6)

How are the boundaries of the 2-D plane strain model determined? Provide and justify the criteria used for such determination.

Response

The boundaries of the 2D plane strain models were placed sufficiently far from the structures to eliminate boundary effects. The vertical boundary (on either side) is at a distance, from the centerline, of more than 5 times the half width of the foundation. The depth is roughly 1 1/2 times the width of the foundation.

The adequacy of the boundaries in the mathematical models (length and depth) was proved by the good correlation of results from the finite element and half-space calculations as described in Section 3.7.1.6 of the PSAR (Page 3.7-3.c). Since the half-space equations are based on boundaries being at an infinite distance, increasing the distances to the boundaries in the finite element model would have no further impact on the results (with the current boundaries).

Amend. 70
Aug. 1982

Question CS220.12 (3.7.2.1)

You used the terms geometrical damping and critical damping. However, in Section C.1.2 of Appendix 3.7A, terms such as interaction damping, radiation damping and internal damping are used. Explain in detail the differences in these terms. If two different terms mean the same thing, use one term consistently in order to avoid confusion.

Response

DAMPING DEFINITIONS FOR THE PSAR

In the equations of motion for the seismic analysis of Nuclear Power Plants, the energy losses are expressed by the damping forces. The energy losses include material damping and hysteretic losses within the structure or in the surrounding soil materials; frictional losses at structural connections, equipment mounts or at soil-structure interfaces; and the radiation of energy away from the structure foundation and into the surrounding soil (radiation damping). For computational expediency the various energy losses are represented in dynamic analysis as equivalent viscous (relative velocity dependent) damping factors. The following define the different terms related to damping used in CRBRP PSAR.

Damping Coefficient

The damping force in the dynamic equation of a one degree of freedom sysem is given by:

$$- c \dot{y} \qquad (1)$$

Where:

c = Mathematical constant - Damping Coefficient

$\dot{y}$ = Relative Velocity

Critical Damping

It is the amount of damping that would completely eliminate vibration in a one degree of freedom system, that is, the smallest amount of damping that will make the system return to the original position.

$$c_{crit} = 2 \sqrt{KM} = 2 M_{\omega} \qquad (2)$$

Where:

K = Stiffness of System

M = Mass

ω = Frequency in Radians Per Second

Damping Ratio, Damping Factor

This is the ratio between the damping coefficient of a one degree of freedom system and the critical damping

$$\beta = \frac{C}{C_{crit}} = \frac{C}{2M\omega} \qquad (3)$$

In experimental determinaticns of damping, the damping ratio is normally measured.

In a one degree of freedom system if the damping ratio is known, the damping coefficient may be calculated by equation (3).

Percent of Critical Damping

This is the damping ratio expressed in percent:

$$D\% = 100\,\frac{C}{C_{crit}} \qquad (4)$$

Table 3.7-2 in the PSAR gives damping values in terms of percent of critical damping.

Damping Matrix

In a multidegree of freedom system the damping coefficients may be expressed in matrix form, by the damping matrix

$$C = \begin{bmatrix} c_{11} & -- & c_{ij} & -- & c_{in} \\ \vdots & & c_{ij} & & \\ c_{ni} & -- & c_{nj} & -- & c_{nn} \end{bmatrix}$$

The coefficients c_{ij} of the damping matrix represent the velocity dependent damping forces developed at coordinate i by a unit velocity imposed at point j.

Modal Damping

Modal Damping is the damping coefficient or the percent of critical damping associated with a particular mode of the modal being analyzed. In modal analysis each mode may have a characteristic or generalized damping which may be expressed as a generalized damping coefficient or generalized damping ratio (or factor). The generalized modal damping coefficient is given by:

$$c_r = \phi_r^T C \phi_r \qquad (5)$$

Where:

c_r = Damping coefficient for mode r

ϕ_r = Mode shape vector for mode r

C = Damping Matrix

The generalized modal damping factor (or ratio) is given by:

$$\beta_r = \frac{\phi_r^T C \phi_r}{2 \omega_r M_r} \qquad (6)$$

Where:

β_r = Generalized damping factor (or ratio) for mode r.

ω_r = Modal frequency for mode r in radians per second

M_r = Generalized mass for mode r

To meet the orthogonality conditions that permit the decoupling of the modal equations the damping matrix must meet certain requirements. However, in modal analysis the damping matrix was not used and damping ratios were defined for the different modes based on the characteristics of the system.

<u>Composite Modal Damping - Combined Modal Damping</u>

When a structure is composed of elements with different damping ratios the modal damping ratio was calculated by a weighted average based on the equation:

$$B_j = \frac{\{\phi_j^T\} \bar{K} \{\phi_j\}}{\{\phi_j^T\} K \{\phi_j\}} \qquad (7)$$

Where:

B_j = Composite Modal Damping for mode j

$\{\phi_j\}$ = Normalized modal vector for the jth mode

K = Assembled stiffness matrix

$\bar{K}$ = Modified stiffness matrix constructed from elements matrices formed by the product of the damping ratio for the element and its stiffness matrix. Damping ratios for elements were based on appropriate Table 3.7-2 values.

In the Nuclear Island model, the damping ratios for the modes of the fixed based were calculated by this method. The damping matrix used in the soil structure interaction of the Nuclear Island is a combination of these fixed base modal damping ratios and the damping coefficients associated with the foundation dampers. The damping matrices are constructed by the computer programs HETHA-VETHA following a formulation similar to that proposed in Reference (5) of the PSAR. Attachment A describes how this reference was used.

Proportional Damping - Raleigh Damping

This is the system damping which is composed of material and interface shear damping. In the seismic analysis of systems by time-history with direct integration of the coupled equations of motion, the operations are simplified by using proportional or Raleigh damping. By this procedure the damping matrix is formulated by a linear combination of the stiffness and mass matrices:

$$C = \alpha |M| + \beta |K| \tag{8}$$

Where:

C = Damping Matrix

$|M|$ = Mass Matrix

$|K|$ = Stiffness Matrix

α & β = Coefficients

α & β are selected for the range of frequencies of interest with the appropriate damping ratio $\underline{c}_{crit}$

α & β are calculated with the following equation:

$$\underline{c}_{crit} = \frac{\alpha}{2\omega} + \frac{\beta\omega}{2}$$

Where:

$\underline{c}_{crit}$ = Damping ratio - (Based on appropriate Table 3.7-2 values)

ω = Frequency (Rads/sec)

Internal Damping - Material Damping

Internal or material damping is the hysteretic damping of the material. For a soil foundation it is the damping characteristic of the soil material. Although the internal damping is not the result of a viscous behavior, it is expressed in terms of an equivalent viscous damping ratio.

Geometric Damping - Radiation Damping

Geometric or Radiation Damping represents the loss of energy that occurs through transmission of elastic wave energy from the footing to infinity in a footing oscillating on an elastic half-space.

This damping may be expressed as a damping ratio or damping coefficient.

Interaction Damping

The total foundation damping is the summation of the soil internal and radiation dampings.

In the CRBRP Nuclear Island analysis the rock internal damping was conservatively disregarded and the interaction damping was made equal to the radiation damping. This damping was expressed as a damping coefficient.

Summary

In system modal analysis, modal dampings in terms of percent of critical were used. The modal dampings are based on the values listed in Table 3.7-2. When elements with different dampings were present in a system, the composite modal dampings were calculated and used.

In systems analyzed by direct integration time-history analysis, proportional (Raleigh) damping was used.

In the analysis of the Nuclear Island (by direct integration) a damping matrix was constructed based on the composite modal dampings of the fixed base structure and the damping coefficients (interaction damping) of the foundation dashpots.

Reference

QCS220.12-1 N. C. Tsai, "Modal Damping for Soil-Structure Interaction," ASCE Journal of the Engineering Mechanics Division, April 1974, pages 323 to 339.

1. DESCRIPTION OF HOW CRBRP HAS USED THE PAPER BY TSAI ON DAMPING

The seismic analysis of CRBRP is performed with the computer programs HETHA (for horizontal motions) and VETHA (for vertical motions).

These programs, developed by Burns and Roe, solve the coupled equations of motion by direct integration allowing for a rigorous solution of the soil-structure interaction problem. The formulation of the equations of motion is similar to that used by Tsai in Reference Q220.12-1.

This approach considers the displacements of the nodes of the structure subjected to an earthquake as composed of two parts: The displacements relative to the base which are expressed in terms of the modes of vibration of the "fixed" base structure and the displacements due to the base motion. The equations of motion are expressed in terms of the mode shapes, modal frequencies and modal dampings of the "fixed" base structure and of the stiffness and damping coefficents of the lumped springs and dashpots that represent the effects of the soil on the structure.

As a result of this formulation the number of coupled equations to be integrated is reduced from N+P to M+P, where N is the number of degrees of freedom of the "fixed" base structure, P the number of degrees of freedom of the base and M the number of modes of the "fixed" base structure selected to represent the structure. For a large structures M is much less than N.

While Tsai's formulation includes only two degrees of freedom (horizontal translation and rocking) the HETHA formulation considers three degrees of freedom per node (horizontal translation, rocking and torsion). VETHA considers one degree of freedom per node (vertical translation). In Ref. 1, Tsai proposed the use of the exact formulation to calculate equivalent modal dampings for a modal analysis of the structures by matching the responses of the exact analysis with those of a modal analysis. In the CRBRP analysis, the coupled equations of motion, similar to equation (15) of Ref. 1 are integrated directly using the acceleration time history of the ground motion as input.

EQUATIONS USED IN HETHA

The following equations were used in the computer program HETHA:

Let X be the "fixed" base displacement of the structure

$$\{X\} = \begin{Bmatrix} \{\delta'\} \\ \{\varphi'\} \\ \{\theta'\} \end{Bmatrix} \qquad (1)$$

Where:

$\{\delta'\}$ = Horizontal displacement in the direction of the input motion, relative to the base.

$\{\varphi'\}$ = Rotation about a horizontal axis perpendicular to the direction of the input motion, relative to the base.

$\{\theta'\}$ = Rotation about a vertical axis relative to the base (torsion)

The equations of motion for a fixed base structure are:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = -\ddot{u}(t)[M]\{\Delta\} \qquad (2)$$

Where $[M]$, $[C]$ and $[K]$ are the mass, damping and stiffness matrices respectively of the fixed base structure

$u(t)$ = Translational ground acceleration (free field)

$$\{\Delta\} = \begin{Bmatrix} \{1\} \\ \{0\} \\ \{0\} \end{Bmatrix}$$

$$[M] = \begin{bmatrix} m_1 \cdots m_N \; I_1 \cdots I_N \; J_1 \cdots J_N \end{bmatrix}$$

Where:

m_i = mass of nodal point i

I_i = mass moment of inertia at nodal point i about a horizontal centroidal axis perpendicular to the direction of the motion

J_i = mass moment of inertia at nodal point i about a vertical centroidal axis (torsional)

Assuming that the fixed base structure has classical normal modes, the normal mode matrix $[\phi]$ satisfies the following conditions:

$$[\phi]^T [M] [\phi] = [I] \qquad (3)$$
$$[\phi]^T [K] [\phi] = [\omega_j^2] \qquad (4)$$
$$[\phi]^T [C] [\phi] = [2\beta_j \omega_j] \qquad (5)$$

Where ω_j are the modal frequencies and β_j the modal fraction of critical damping for the fixed base structure.

For the same structure with flexible foundation, the displacements of the structure relative to the ground motion are given by:

$$\delta_i = \delta'_i + u_b + \psi h_i + \theta e_i \qquad (6)$$
$$\varphi_i = \varphi'_i + \psi \qquad (7)$$
$$\theta_i = \theta'_i + \theta \qquad (8)$$
$$1 > i > N$$

Where:

δ_i = Horizontal displacement of point i in the structure in the direction of the input motion, relative to the ground motion.

γ_i = Rotation at point i in the structure about a horizontal axis perpendicular to the direction of the input motion relative to the ground motion.

θ_i = Rotation at point i in the structure about a vertical axis, relative to the ground motion.

y_b = Horizontal translation of the base mat.

ψ = Rotation of the base mat about a horizontal axis perpendicular to the input motion. The axis passes through the center of rigidity of the base mat.

θ = Rotation of the base mat about a vertical axis. The axis passes through the center of rigidity of the base mat.

h_i = Height of point i in the structure relative to the base mat.

e_i = Horizontal distance from point i in the structure to the vertical axis of rotation of the base mat, measured perpendicular to the direction of the input motion.

Define

$$\{y\} = \begin{Bmatrix} \{\delta\} \\ \{\varphi\} \\ \{\theta\} \end{Bmatrix} \tag{9}$$

With the equations of force equilibrium at the base written to satisfy the boundary condition, the equations of motion of soil structure interaction are

$$\begin{aligned}
&[M](\{\ddot{y}\} + \ddot{u}\{\Delta\}) + [C]\{\dot{x}\} + [K]\{x\} = \{0\} \\
&m_b(\ddot{y}_b + \ddot{u}) + c_x \dot{y}_b + K_x y_b = -\sum_{i=1}^{N} m_i (\ddot{s}_i + \ddot{u}) \\
&I_b \ddot{\psi} + c_\psi \dot{\psi} + K_\psi \psi = -\sum_{i=1}^{N} m_i h_i (\ddot{\delta}_i + \ddot{u}) - \sum_{i=1}^{N} I_i \ddot{\varphi}_i \\
&J_b \ddot{\theta} + c_\theta \dot{\theta} + K_\theta \theta = \sum_{i=1}^{N} m_i e_i (\ddot{\delta}_i + \ddot{u}) - \sum_{i=1}^{N} J_i \ddot{\theta}_i
\end{aligned} \tag{10}$$

Where: m_b = Mass of Mat

I_b - Mass Moment of Inertia of Mat about the centroidal horizontal axis perpendicular to the direction of the input motion

J_b - Mass Moment of Inertia of Mat about a centroidal vertical axis

I_i = Mass moment of inertia at point i about a centroidal horizontal axis perpendicular to the direction of the input motion

J_i = Mass moment of inertia at point i about a centroidal vertical axis

Equations (10) can be rearranged to give:

$$
\left[\begin{array}{c|ccc}
[M] & & & \\
\hline
 & m_b & & \\
 & & I_b & \\
 & & & J_b
\end{array}\right]
\begin{Bmatrix} \{\ddot{y}\} \\ \ddot{y}_b \\ \ddot{\psi} \\ \ddot{\theta} \end{Bmatrix} +
$$

$$
\left[\begin{array}{c|ccc}
[C] & -[C]\{\Delta\} & -[C]\{\Delta h\} & -[C]\{\Delta e\} \\
\hline
-\{\Delta\}^T[C] & (C_x + \{\Delta\}^T[C]\{\Delta\}) & \{\Delta\}^T[C]\{\Delta h\} & \{\Delta\}^T[C]\{\Delta e\} \\
-\{\Delta h\}^T[C] & \{\Delta h\}^T[C]\{\Delta\} & (C_\psi + \{\Delta h\}^T[C]\{\Delta h\}) & \{\Delta h\}^T[C]\{\Delta e\} \\
-\{\Delta e\}^T[C] & \{\Delta e\}^T[C]\{\Delta\} & \{\Delta e\}^T[C]\{\Delta h\} & (C_\theta + \{\Delta e\}^T[C]\{\Delta e\})
\end{array}\right]
\begin{Bmatrix} \{\dot{y}\} \\ \dot{y}_b \\ \dot{\psi} \\ \dot{\theta} \end{Bmatrix} +
$$

$$
\left[\begin{array}{c|ccc}
[K] & -[K]\{\Delta\} & -[K]\{\Delta h\} & -[K]\{\Delta e\} \\
\hline
-\{\Delta\}^T[K] & (K_x - \{\Delta\}^T[K]\{\Delta\}) & \{\Delta\}^T[K]\{\Delta h\} & \{\Delta\}^T[K]\{\Delta e\} \\
-\{\Delta h\}^T[K] & \{\Delta h\}^T[K]\{\Delta\} & (K_\psi - \{\Delta h\}^T[K]\{\Delta h\}) & \{\Delta h\}^T[K]\{\Delta e\} \\
-\{\Delta e\}^T[K] & \{\Delta e\}^T[K]\{\Delta\} & \{\Delta e\}^T[K]\{\Delta h\} & (K_\theta + \{\Delta e\}^T[K]\{\Delta e\}
\end{array}\right]
\begin{Bmatrix} \{y\} \\ y_b \\ \psi \\ \theta \end{Bmatrix} +
$$

$$
= \begin{Bmatrix} [M]\{\Delta\} \\ +m_b \\ 0 \\ 0 \end{Bmatrix} \ddot{u} \qquad (11)
$$

where:

$$
\{\Delta\} = \begin{Bmatrix} \{1\} \\ \{0\} \\ \{0\} \end{Bmatrix} \qquad
\{\Delta h\} = \begin{Bmatrix} \{h_i\} \\ \{1\} \\ \{0\} \end{Bmatrix} \qquad
\{\Delta e\} = \begin{Bmatrix} \{e_i\} \\ \{0\} \\ \{1\} \end{Bmatrix}
$$

Define the following transformation:

$$\begin{Bmatrix} \{y\} \\ y_b \\ \psi \\ \theta \end{Bmatrix} = \left[\begin{array}{c|ccc} [\phi] & & & \\ \hline & 1/\sqrt{m_b} & & \\ & & 1/\sqrt{I_b} & \\ & & & 1/\sqrt{J_b} \end{array}\right] \begin{Bmatrix} r_1 \\ \vdots \\ \vdots \\ r_{M+3} \end{Bmatrix} \qquad (12)$$

$$\{V\} = [A]\{r\} \qquad (13)$$

Where M is number of structural modes and let

$$\begin{aligned} \{\Gamma\} &= (\phi)^T\ (M)\{\Delta\} \\ \{\Gamma_1\} &= (\phi)^T\ (M)\{\Delta'\} \\ \{\Gamma_2\} &= (\phi)^T\ (M)\{\Delta''\} \\ \{\Gamma_3\} &= (\phi)^T\ (M)\{\Delta h\} \\ \{\Gamma_4\} &= (\phi)^T\ (M)\{\Delta e\} \end{aligned} \qquad (14)$$

Where: $\{\Delta'\} = \begin{Bmatrix} \{0\} \\ \{1\} \\ \{0\} \end{Bmatrix} \qquad \{\Delta''\} = \begin{Bmatrix} \{0\} \\ \{0\} \\ \{1\} \end{Bmatrix}$

Premultiply equation (11) by $[A]^T$ and substitute equation (13). Equation (11) becomes:

$$\{\ddot{r}\} + [\bar{C}]\{\dot{r}\} + [\bar{K}]\{r\} = -\{\bar{\Gamma}\}\ddot{u} \qquad (15)$$

let $\quad B_j = 2\beta_j\omega_j$

$$[\bar{C}] = \left[\begin{array}{c|ccc} [B_j] & -[B_j]\{\Gamma\}/\sqrt{m_b} & -[B_j]\{\Gamma_3\}/\sqrt{I_b} & -[B_j]\{\Gamma_4\}/\sqrt{J_b} \\ \hline \dfrac{-\{\Gamma\}^T[B_j]}{\sqrt{mb}} & (C_x + \sum_{j=1}^{M} B_j\Gamma_j^2)/m_b & \sum_{j=1}^{M} B_j\Gamma_j\Gamma_{3j}/\sqrt{m_bI_b} & \sum_{j=1}^{M} B_j\Gamma_j\Gamma_{4j}/\sqrt{m_bJ_b} \\ \dfrac{-\{\Gamma_3\}^T[B_j]}{mb} & \sum_{j=1}^{M} B_j\Gamma_j\Gamma_{3j}/\sqrt{m_bI_b} & (C_\psi + \sum_{j=1}^{M} B_j\Gamma_{3j}^2)/I_b & \sum_{j=1}^{M} B_j\Gamma_{3j}\Gamma_{4j}/\sqrt{I_bJ_b} \\ \dfrac{-\{\Gamma_4\}[B_j]}{\sqrt{J_b}} & \sum_{j=1}^{M} B_j\Gamma_j\Gamma_{4j}/\sqrt{m_bJ_b} & \sum_{j=1}^{M} B_j\Gamma_{3j}\Gamma_{4j}/\sqrt{I_bJ_b} & (C_\theta + \sum_{j=1}^{M} B_j\Gamma_{4j}^2)/J_b \end{array}\right] \qquad (16)$$

Matrix $[\bar{K}]$ is identical to $[\bar{C}]$ when $B_j = W_j^2$, $C_x = K_x$, and $C_\psi = K_\psi$ and $C_\theta = K_\theta$ (17)

$$\{\bar{r}\} = \begin{Bmatrix} \frac{\{r\}}{\sqrt{m_b}} \\ 0 \\ 0 \end{Bmatrix}$$

$\{r\}$ in eq.(13) is an $N+2$ by 1 column vector. Considering only the first M modes of the fixed base structure, $\{r\}$ becomes a $M+3$ by 1 column vector.

Equation (15) is solved by Wilson's (2) step by step integration. Let $\{r_t\}$, $\{\dot{r}_t\}$ and $\{\ddot{r}_t\}$ be known as vectors at time t, the acceleration vector $\{\ddot{r}_{t+\tau}\}$ is solved by the equation:

$$[M]\{\ddot{r}_{t+\tau}\} + [C]\{\dot{r}_{t+\tau}\} + [K]\{r_{t+\tau}\} = \{\tilde{R}_{t+\tau}\} \quad (18)$$

Where velocity and acceleration are replaced by:

$$\{\dot{r}_{t+\tau}\} = \frac{3}{\tau}(\{r_{t+\tau}\} - \{r_t\}) - 2\{\dot{r}_t\} - \frac{\tau}{2}\{\ddot{r}_t\} \quad (19)$$

$$\{\ddot{r}_{t+\tau}\} = \frac{6}{\tau^2}(\{r_{t+\tau}\} - \{r_t\}) - \frac{6}{\tau}\{\dot{r}_t\} - 2\{\ddot{r}_t\} \quad (20)$$

$$\{\tilde{R}_{t+\tau}\} = \{R_t\} + \frac{\tau}{\Delta t}(\{R_{t+\Delta t}\} - \{R_t\})$$

$$\{R_t\} = -\{\bar{r}\}\ddot{u}$$

$$\tau = \alpha \times \Delta t \quad (21)$$

Velocity and acceleration vectors at $t+\tau$ are obtained by equations (19) and (20).

The acceleration, velocity and displacement at $t+\Delta t$ are given by:

$$\{\ddot{r}_{t+\Delta t}\} = (1-\frac{1}{\alpha})\{\ddot{r}_t\} + \frac{1}{\alpha}\{\ddot{r}_{t+\tau}\}$$

$$\{\dot{r}_{t+\Delta t}\} = \{\dot{r}_t\} + \frac{\Delta t}{2}(\{\ddot{r}_t\} + \{\ddot{r}_{t+\Delta t}\})$$

$$\{r_{t+\Delta t}\} = \{r_t\} + \Delta t\{\dot{r}_t\} + \frac{\Delta t^2}{6}(\{\ddot{r}_{t+\Delta t}\} + 2\{\ddot{r}_t\}) \quad (22)$$

The displacement and accelerations (r) are combined to obtain nodal responses according to equation (13).

Question CS220.13 (3.7.2.1)

At top of page 3.7-6a, it is indicated that in the model the dome of the steel containment has been idealized using equivalent springs which account for the "breathing" of the dome during a vertical vibration. Explain clearly how through such a lump-mass model "breathing" of the dome can be taken into consideration. Define "breathing".

Response:

"Breathing" refers to shell type of vibration of the dome. To account for this effect, a stiffness matrix of the dome with cross-coupling terms was derived from an axisymmetrical shell model of the dome, using the KALNINS computer program. The equivalent springs represent the terms in the stiffness matrix.

Question CS220.14 (3.7.2.1)

Two computer programs: HETHA and VETHA are mentioned. Indicate if these two computer programs are validated in accordance with the procedure described in SRP Section 3.8.1 (P.3.8.1-10).

Response

The computer programs HETHA and VETHA have been validated consistent with the requirements of SRP Section 3.8.1 (Page 3.8.1-10), item ii. A description of the programs is in PSAR Appendix A pages A169 to A182.

Question CS220.15

On page 3.7-8, it is stated that you perform a non-linear history analysis. Provide more details of such an analysis.

Response

The nonlinear seismic analysis of reactor systems was performed on the Primary Control Rod System (PCRS) since this system was determined to contain significant nonlinearities. The description of this analysis is given in Section 3.7.3.15.3 of the PSAR.

Question CS220.16 (3.7.2.6)

On page 3.7-9b, it is stated that two methods of combining the seven spectra, one by square root of sum of squares and the other by absolute sum. Indicate the conditions under which each of the methods will be used.

Response

First method:

Each of the seven spectra are applied independently to find the final effects (stresses, deflections, etc) and the resultant stresses are combined as shown by equations (8), (9), and (10) on page 3.7-A-B3 of the PSAR. These equations add the total effects of the north-south, east-west and vertical earthquakes by the square root of the sum of the squares.

Second method:

Instead of using each of the seven spectra individually, the seven spectra are combined into three which correspond respectively to the north-south, east-west, and vertical directions, using equations (11), (12) and (13) on page 3.7-A-B3 of the PSAR. The final effects obtained with these combined spectra are added by absolute values.

The selection of one or other method is optional to the designer.

Question CS220.17 (3.7.2.13)

The discussion on overturning of seismic Category I structure appears to be for a structure on a single foundation mat not on a combined foundation. Indicate what your consideration will be for the overturning of structures on a combined foundation mat. Express with the help of a figure the location of H_1 and the distance of h_1 in the equation on page 3.7-10a and the basis of their determination.

Response:

Figure 3.7-19 in the PSAR shows the single foundation mat of the Nuclear Island structures. In the seismic mathematical model, the rock-structure interaction was represented by lumped springs. From the seismic analysis, the maximum foundation spring forces, (translational and rotational) were calculated. These forces were applied in verifying the stability of the structure. Thus, in the equation on page 3.7-10a in the PSAR:

$$M_{ot1} = Mr_1 + h_1 H_1$$

Mr_1 is the maximum reaction in the rotational foundation spring in the north-south direction. H_1 is the maximum reaction of the north-south translational spring, and h_1 is the distance from the force H_1 to the bottom of the foundation mat. h_1 was conservatively made equal to the foundation mat thickness. Mot_1 is the total overturning movement due to the north-south earthquake.

Figure QCS220.17-1 shows a sketch of the forces.

Question CS220.18 (3.7.2.14)

In this section you discussed the analysis procedure for damping. Indicate if there is any difference between what you described in this Section and that in Section 3.7.2.1.1 on pages 3.7-5 and 3.7-6. Explain in mathematical forms the following: composite damping, modal damping, proportional damping, and their relationships to the critical damping values as specified in Table 3.7.2, together with the conditions under which they are used.

Response:

See the Response to Question QCS220.12.

Question CS 220.19 (3.7.3.10)

In this section it is stated that the response spectra produced will be widened by ±10% by frequency to account for uncertainties in the structural model and imput. However, in Section 3.7.1.6 on page 3.7-3, it is stated that the response spectra will be widened by ±15% in frequency. Indicate which percentage of widening is actually used and if its use is in conformance with SRP Section 3.7.2 criteria.

Response

The design response spectra is the envelope of response spectra produced for the upper and lower bound properties of the rock (soil) properties. Before enveloping, the spectral peaks were widened by ±10%. The ±15% indicated in Section 3.7.1.6, page 3.7-3 has been revised to ±10%. The ±10% is consistent with SRP which requires a minimum of 10%. In addition, CRBRP envelopes the results of upper and lower bound of rock (soil) properties.

Question CS 220.20

In Section 6.1 on page 3.7-A.8, your definition of significant dynamic modes is not consistent with that in SRP Section 3.7.2 and should be revised. Further, the sentence before the last sentence in the third paragraph stated that different response spectra will be applied for the particular support location. An explanation should be given to this statement.

Response

The definition of significant dynamic modes has been revised, as shown marked on PSAR Page 3.7-A.8.

The different response spectra are the spectra derived and applicable to the various supports of structural systems (such as piping) which are not at the same location. The spectrum at the first support may not be the same as the spectrum at an intermediate support or at the last support in this case, all the different support response spectra are superimposed to yield an envelope response spectrum for input to the response spectrum analysis.

Amend. 70
Aug. 1982

Question CS220.21

In Sections 8.1.1.1 and 8.1.1.2 on page 3.7-A, the listed load combinations contain the term "operating." Define specifically what are the loads included in this term.

Response:

The loading components included in the term "Operating" are given in its definition in Section 7.1.1, Page 3.7-A.11 of the PSAR.

Question CS220.22(a)

There are a number of misprints, unclear statements and typographical errors which need your correction and/or clarification.

o Section 3.7.1.6 page 3.7-3C misprints in both items 2) and 3).

Response:

The word "p[rocedure" should be "procedure".

The word "axiaymmetrical" should be "axisymmetrical".

PSAR Page 3.7-3C has been revised.

Question 220.22(b)

The equation for the damping values β_r in Section 3.7.2.1.1 is in error. (Probably misprint)

Response:

The correct equation is:

$$\bar{\beta}_r = \frac{\{\phi\}^T(\bar{K})\{\phi\}}{\{\phi\}^T(K)\{\phi\}}$$

There is a misprint in the PSAR. $(\bar{K})$ in denominator should have been (K).

PSAR page 3.7-5 has been revised.

Question CS220.22(c)

In Section 3.7.2.6.1 on page 3.7-9, it is stated that each node has three degrees of freedom in the horizontal directions. This is a wrong statement, since each node should have six degrees of freedom. A correction of this statement should be made.

Response:

The first paragraph of Section 3.7.2.6.1 has been revised to clarify the statement.

Question CS 220.22 (d, i)

There are a number of misprints, unclear statements and typographical errors which need your correction and/or clarification.

d) Section 3.7.2.7 on page 3.7-9b, the last three sentences need some correction or clarification in order to be understandable.

Response

d) A missing word has been properly inserted in Section 3.7.2.7, Page 3.7-9b of the PSAR. This will clarify the meaning of the last three sentences.

Question CS220.22(e)

Section 3.7.2.14, misprint in the fourth paragraph.

Response:

The word "requesented" should be "represented". PSAR page 3.7-11 has been revised.

Question CS220.22(f)

Section 3.7.3.1, in the last sentence, is the word "assured" used for "assumed"?

Response:

The word "assured" should be "assumed". PSAR page 3.7-11a has been revised.

Question CS220.22(g)

Section 3.7.4.2, next to last paragraph under 1, on page 3.7-17, the word "Time-History".

Response:

The word "Time-hisotry" should be "time-history". PSAR page 3.7-17 has been revised.

Question CS220.22(h)

In Table 3.7-5, you used a poisson ratio of 0.3 for both limestone and siltstone. Indicate how this value is obtained and why it is the same for both.

Response:

The derivation of Poisson's Ratio for siltstone and limestone has been based on detailed analysis of a significant number of geophysical measurements as obtained from cross-hole, up-hole and continuous velocity methods. Similar ranges were obtained by the different methods for each rock type, resulting in the selection of 0.3 as the design value. Details are provided in Section 2.5.4.2.2 of the PSAR.

Question CS 220.22 (i)

There are a number of misprints, unclear statements and typographical errors which need your correction and/or clarification

i) The Figures 3.7.17D and 3.7.18 have the same title, but it is not clear how they are related. Clarify.

Response

i) The title of Figure 3.7.17D has been changed to "Schematic of Reactor System Finite Element Model." In addition, an introductory sentence has been added to Section 3.7.3.15.2 to identify this figure.

Question CS220.22(j)

j) Appendix 3.7 in "Attachment A", the equation for a simplified analysis appears to have incorrect units. Also the transformation equations appear to be incorrect.

Response:

The corrections to the errors in the transformation operations sited above appear in the revised PSAR page 3.7-A-A1. The errors were primarily typographical in nature.

With regard to the simplified analysis equation, (Equation 19) it should be noted that the spectral accelerations are given in "g" units. The multiplication of the weight by "g" gives a force. Therefore, the equation is dimensionally correct. A revision will be made to clarify the units for "A"s.

Question CS220.23 (3.8.2.1)

It is stated to the effect that the steel shell in the lower portion of the containment structure is sandwiched between two concentric concrete walls and neither of the two concrete walls are considered to be part of the containment steel. However, the outside concrete wall is indicated to be designed to prevent the buckling of the steel shell. Indicate what are the design criteria for the two concrete walls, especially the outside concrete wall, and which ACI Code will be used in your design.

Response:

The walls are designed to resist all normal, seismic and accident loads in accordance with the load combinations specified in PSAR Section 3.8.3.3. The design of the walls is done in accordance with ACI 349.

Question CS 220.24

The containment description should include basic shell thickness and state if the shell is stiffened. The containment vent and purge system should be mentioned in the list of components.

Response

Horizontal ring stiffeners are provided at elevations 856 and 839. In addition the crane girder also functions as a horizontal ring stiffener in the elevation 870 to 890.

The PSAR has been augmented with the requested information in 3.8.2.1 and also Figure 3.8-3.

The containment vent and purge systems are features provided to manage the hypothetical event beyond the design basis. These systems are discussed in CRBRP 3, Vol. 2, Figure 1-1. Only the penetrations for the vent and purge systems are a part of the containment boundary and these penetrations are designed to the same criteria as all other penetrations. The vent and purge system is not a part of the containment system and therefore should not be included in the list of containment system components.

Question CS220.25

On page 3.8-1, it is stated that ASME Section III Division 1, 1974 Edition with Addenda through winter 1974 and ASME Section III Division 2, 1975 Edition will be used for the design of the steel containment and the steel lined concrete containment foundation mat respectively. Indicate what will be the effect on the design of the latest editions of the ASME Section III Division 1 and 2 including Code Case N-284 (1980) are used.

Response

The PSAR design was performed to the requirements of the 1974 Code edition specified in the design specifications. The specific criteria related to buckling are described in the PSAR Appendix 3.8-A. The intent of these criteria is similar to the Code Case N-284 criteria, in that these address buckling modes, provide capacity reduction factors and factors of safety, and similar interaction equations for buckling. A significant reanalysis would be required to demonstrate that the Containment Vessel meets the requirements of the new Code and Code Case N-284, however, the applicant has compared the PSAR to the 1980 ASME Code, and has evaluated the significance of the changes. Several of these changes are considered to be of sufficient significance to require additional study. This comparison will be provided by July 15, 1982, and the additional study of the significant changes will be provided by August 30, 1982.

The applicant believes that the intent of N-284 and the 1980 ASME Code has been implemented by the PSAR and the PSAR Appendix 3.8-A, the vessel design is adequate and safe, and that no analysis to the 1980 Code or to Code Case N-284 is necessary.

Question CS220.26 (2.8.2.2.2)

It is stated that potential corrosion of the portion of steel containment embedded in concrete as a result of concrete cracking is precluded due to fact that there is a minimum of 22 inches of concrete embedment and the cracking under the worst of cases is minimal. Indicate what size of cracks is expected as a result of the containment structural integrity test. Note that these cracks will terminate at the steel containment shell.

Response:

The size of possible cracks in the 22 inch concrete wall around the containment vessel for a test pressure of 11.5 psi has been calculated. It has been assumed conservatively that the internal pressure acts directly on the containment shell and that the load is shared by the steel shell and the 22 inch concrete wall in accordance with their relative stiffnesses without assistance from the interior 36 inch concrete wall. The calculations are based on the hoop stresses developed in the concrete by the 11.5 psig test pressure. The analysis method of Reference QCS220.26-1 (Section 10.4.3) was used.

The calculated crack width is 0.009 inch. In accordance with ACI Committee 224 (Reference QCS220.26-2) the following are acceptable crack widths in concrete:

In dry air:	0.016 inch
Humidity, moist air, soil:	0.012 inch

The calculated crack width is below the applicable limit which is 0.016 inch. It should be noted, that the stresses in the steel shell and reinforcing steel are below yield. Therefore, when the vessel is depressurized and the tensile stresses relieved, the cracks will close. Based on the above discussion there is no concern for corrosion in the steel shell.

References:

QCS220.26-1 Park and Pauley, "Reinforced Concrete Structures", John Wiley, 1975.

QCS220.26-2 ACI Committee 224, "Control of Cracking in Concrete Structures", Journal ACI, Vol. 69, No. 12, December 1972 pp. 717-753.

Question CS 220.27

The design temperature of 250°F must not apply to the complete containment including that portion embedded in concrete. The PSAR should define the design temperature distribution for the complete containment.

Also the Symbol "W" used in Table 3.8-1 is not defined in test.

Response

The Design Temperature of 250°F is a conservative value, well above the maximum temperature of the shell calculated under DBA. For Design Conditions (Load Combinations 4 to 9, Table 3.8-1) the material properties of the steel shell were based on 250°F. Secondary stress (thermal) verification is not required under Design Conditions (PSAR Table 3.8-3). Therefore, temperature distributions are not required. Thermal buckling was verified based on the temperature distribution (axisymmetrical) given in Figure 6.2.11. This temperature distribution accounts for the effects of the insulation blanket between Elevations 825 feet and 816.0 feet and the embedment in concrete below Elevation 816.0 feet.

The definition of "W" wind level was erroneously omitted from the PSAR. "W" is now defined in the modified PSAR 3.8.2.3.1 attached.

Question CS 220.28

The applicant should substantiate the statement that the containment will not be subjected to non-axisymmetric temperature distributions above the operating floor.

Response

For the postulated containment vessel Design Basis Accident (DBA) the containment vessel would be heated by convection from the hot gasses from Cell 102A entering the spaces above the operating floor, and by conduction through the concrete. These gasses would enter this above operating floor space through potential paths that are widely distributed. One potential path is Cell 105U to Cells 109 and 113; another is Cell 105H to Cells 109 and 113; and another is Cell 105H to Cell 110 on the opposite side. (See Figures 1.2-14 through 1.2-16 of the PSAR). All of these paths are very unlikely and contain fire doors and other inpediments to the fire passage.

As shown in PSAR Section 6.2 the peak gas temperature above the operating floor is only 140^{o}F which occurs only after a long time and at a very slow rate. This is only a 70^{o}F rise above the normal operating temperature (and normal shell temperature); thus, the magnitude of the temperature variation could only be a portion of the 70^{o} which would create negligable stress in the shell, even if the heating source was localized.

It should be noted also that the design basis accident sodium fire analysis, which is used to postulate containment temperatures, assumes that hot gasses are transmitted directly to the space above the operating floor from the fire in Cell 102A. This assumption maximizes the rate and total amount of heat transferred into the area above the operating floor without taking credit for the heat losses into the concrete and equipment along the actual gas flow paths. This is a conservative assumption which cannot actually occur and therefore adds even more conservatism to the design.

Question CS220.29 (3.8.2.4)

Indicate if there are any mechanical connections between the confinement structure and the containment above the operating floor that could transmit mechanical loads. Can the relative displacement between the confinement structure and containment become large enough to allow contact between the containment and components attached to the confinement structure (such as the partition supports)?

Response:

There are no mechanical connections between the confinement structure and the containment above the operating floor that could transmit mechanical loads.

Further, the gaps between the containment vessel and components attached to the Confinement (such as partition supports) will be established to prevent contact under any loads, including Thermal Margin Beyond the Design Basis (TMBDB) accidents.

Question CS 220.30

The ultimate capacity of the steel containment should be addressed.

Response

The ultimate capacity of the steel containment is addressed in the document "CRBRP-3, Hypothetical Core Distruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base." A tabulation of the allowable pressures for different temperatures of the material is given in Table 3-10 of the above document. These pressures were calculated with primary membrane stress limits for Service Limit Level D given in NUREG-0800 (SRP 3.8.2), with yield and ultimate strength values from the Nuclear Systems Materials Handbook (TID-26666). The above allowable pressures are conservative since the thickness of the cylindrical portion of the containment has increased subsequent to the calculations due to other design considerations.

Question CS 220.31 (3.8.3.1.3)

On page 3.8-10, it is stated that the interior surfaces of the cells are lined with carbon steel plates with the lower portion of the plate designed to contain hot sodium spills. Indicate the difference in the design of the lower and upper portion of the cell lines.

Response

The design of the cell liner system does not distinguish between wetted (lower) and unwetted (upper) zones of the lined cell. As identified in PSAR Section 3.8-B, paragraph 3.1.1.9, the cell liners are designed to withstand large sodium spills with Na spill temperatures up to 1015°F consistent with their application. PSAR paragraph 3.8.3.1.3 has been revised to be consistent with the design of the cell liner system.

Question CS 220.32 (3.8.3.2.1)

It is stated that concrete internal structures will be designed in accordance with ACI 318-77. Since ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures" is specifically for the design of such structures and has been endorsed by NRC in Regulatory guide 1.142, use of this code is required.

Response

Section 3.8.3.2.1 of the PSAR will be updated to state the design of the internal structures complies with the requirements of ACI 349, "Code Requirements for Nuclear Related Concrete Structures," as endorsed in Regulatory Guide 1.142.

Question CS 220.33 (3.8.3.4)

The general structural analysis procedure using the strip method is not totally clear. If possible, the method should be referenced to the ACI-349(76) code. If not, more detail is needed on how the interaction of surrounding cells will be handled when analyzing individual cells. Can significant additive moments be introduced from adjacent cells at a common juncture? Will the method described take into account the high tensile loads developed on the diagonals of two-way slabs near the corners?

Response

In the analysis of concrete structures such as the cells within the RCB, a normal procedure is to consider strips of unit width and to analyze the strips as frames. The strips are taken along sections of the structure and include interaction of walls and slabs of adjacent cells. The design criteria is based on ACI-349.

The high tensile loads developed on the diagonals of two-way slabs near the corners apply only to exterior slabs framed by spandrel beams and does not apply to slabs integrally framed into exterior walls and hence is not applicable to the Reactor Containment Building. Section 13.5 of ACI 349(76) provides minimum reinforcement requirements for this condition. The reinforcement steel provided in the slabs exceeds, by far, the minimum requirement for slabs on spandrel beams.

Question CS 220.34 (a)

There are a number of misprints, unclear statements and typographical errors which need your correction and/or clarification.

a) Figure 3.8-9 should label elements discussed in the text (3.8.3.1.1). Details of concrete reinforcing are needed to evaluate the support ledge.

Response

Revised PSAR Figure 3.8-9 labels elements discussed in the text (3.8.3.1.1). An additional sketch, showing the reinforcing bars in the reactor cavity at the ledge area has been included.

Question CS 220.34 (b)

Is the load of 50,000 kips mentioned in Section 3.8.3.3.4 to be evenly distributed around the support ledge?

Response

The load of 50,000 kips on the Reactor Support Ledge, stated in Section 3.8.3.3.4, has been revised. The revised load is given in terms of time-histories for vertical and toroidal loads. This time-histories are shown in CRBRP-3.

These loads are axisymmetrical and therefore are applied uniformly around the support ledge.

Question CS 220.34 (c)

The text at the top of page 3.8-16 is generally confusing. In particular, the text seems to negate the need for having both cases 10 and 11. The PSAR should delineate how the appropriate dynamic load factor will be determined. Load combination 10 and 11 need more justification for not including T_a unless A is meant to include thermal effects. In that case, the definition of A in 3.8.3.3.1.4 needs to be changed.

Response

For load combinations 6, 7 and 8 in some cases dynamic analysis based on a force time-history input was performed. In other cases, dynamic load factors are calculated using the procedures described in Section 3.5.4.6 of the PSAR which are consistent with the requirements of SRP, Section 3.5.3.II.B.2.

Load combinations 10 and 11 were intended to cover the cases involving margin events beyond the design basis.

Load combination 11 will be deleted since it is already covered in the last paragraph of 3.8.3.3.10.1.B that states: "Both cases of L having its full value of being completely absent will be checked for." Also, the definition of A in 3.8.3.3.1.4 will be revised to state: "A.....force on the structures due to third level design margin requirement (SMBDB)."

Under SMBDB conditions the thermal conditions are represented by To (Ta = To).

The dynamic forces under SMBDB conditions (load "A") are defined as the time histories of the vertical and toroidal moments presented in CRBRP - 3. The effects on the reactor vessel support ledge have been calculated by dynamic analysis.

PSAR Sections 3.8.3.3.1.4 and 3.8.3.3.10.1 will be updated to include the above revised design information.

Question CS220.34 (d)

In combinations (4) and (8) inclusive in Section 3.8.3.3.10.2.B, are thermal loads to be neglected when it can be shown that they are secondary and self-limiting in nature and or or where the material is ductile?

Response:

The second paragraph of Section 3.8.3.3.10.2.B of the PSAR will be updated.

Question CS 220.34(e)

In Section 3.8.3.4, on page 3.8-18, the figure numbrs referenced are incorrect.

Response

The referenced figure numbers in PSAR Section 3.8.3.4, page 3.8-18, have been revised.

Question CS 220.34(f)

In Section 3.8.3.5.2, more description is needed of the "energy absorption check."

Response

Ductility ratio, u, is a measure of the capacity of a structure to absorb energy in the plastic range. Energy absorption is satisfied when the calculated value of the required ductility ratio is less than the allowable ductility ratio for the material under a specific loading condition.

PSAR Section 3.8.3.5.2 will be updated to include the above design information.

Question CS 220.34(g)

In Section 3.8.3.7, if internal structures that are designed to hold more than 10 psig pressure are not to be tested at 1.15 times their design pressure, provide justification for not performing such tests.

Response

Pressure testing of the lined cells is not a requirement since the cells do not perform a containment function. A sodium spill accident within a cell will cause heating of the cell atmosphere and structure and a pressure buildup, but there are no requirements restricting cell atmospheric leakage. The structural design of the concrete walls of the cell, per ACI 349, ensures structural integrity under the sodium spill accident conditions.

The maximum pressure on the containment vessel under the most severe accident condition will not exceed 10 psi. A detailed description of the containment System is provided in PSAR Section 6.2.

Question CS 220.34 (h)

In Section 3.8.4.4.1, how are equivalent static loads obtained?

Response

Equivalent static loads are obtained by equation

$$P = C\,q\,A$$

where P = equivalent static load in lbs

C = drag or lift coefficient

q = dynamic pressure in lb/ft^2

A = exposed area in sq. ft.

The procedure in ANSI A58.1 is followed. PSAR Section 3.8.4.4.1 will be revised to include the above design information.

Question CS 220.34 (I)

The Section 3.8.2.6 and 3.8.2.7 are missing and should be provided. It appears that your Section 3.8.2.5 should be revised.

Response

PSAR Section 3.8.2.5 has been totally revised and PSAR Sections 3.8.2.6 and 3.8.2.7 have been added.

Question CS220.35

a) Code Case N-284 (1980) should be referenced and applied as applicable.

b) The abscissa on Figures 3.8A-1, 3.8A-4, and 3.8A-6 should be labeled R/t and not R/1.

c) Both quadratic and linear interaction curves are used. Most authors recommend using linear interaction curves. Are the nonlinear interaction curves conservative?

d) The R/t range for the containment shell is in a borderline region where either elastic or plastic buckling could occur. For fabricated shells of this type, imperfections can greatly influence the elastic buckling loads. Also, plastic buckling can be influenced by large residual stresses that can be present. For these reasons the factors of safety given in Table 3.8A-2 seem to be low. Please justify these factors.

e) How will buckling be evaluated for dynamic loads?

Response:

a) See answer to Question 220.25. Code Case N-284 addresses contianment shell buckling. This non-mandatory Code Case is a recent document which was not in existence at the time when the Containment Vessel design was initiated. The Project recognized the need for specific buckling criteria and an appropriate criterion was developed which is described in PSAR Appendix 3.8-A. While the quantitative effect on the design of using Code Case N-284 has not been determined, the intent of the buckling criteria used by the Project is similar to the Code Case, in that the criteria address buckling modes, provide capacity reduction factors and factors of safety, and similar interaction equations for buckling.

b) The typographical errors on Figures 3.8A-1, 3.8A-4 and 3.8A-6 have been corrected.

c) The selection of quadratic or linear interaction curves is dictated by loading combination. For example, interaction curves for loading combinations involving torsion are quadratic because modes shapes are dissimilar. Quadratic interaction curves are used in Code Case N-284. In the design buckling criteria, for each loading combination the shape of the nonlinear interaction curve is identical to that given in a widely used shell design document, "Structural Analysis of Shells" by E. J. Baker, L. Kovalesky and F. L. Rish, McGraw-Hill, 1972. Therefore, it is concluded that each interaction curve is conservative.

d) The factors of safety given in Table 3.8A-2 are based on the requirements that the effects of initial imperfections and plasticity are adequately considered in the calculation of critical loads. These effects are accounted for by the coefficients C, K, and H in Figures 3.8A-1 to 3.8A-10. The effects of residual stresses on predicted critical loads are negligible for ring-stiffened cylinders subjected to axial compression (the most severe loading condition).

e) Buckling is evaluated for dynamic loads by comparing the peak dynamic stresses with the static critical (allowable) stresses.

Question CS220.36 (a) (Appendix 3.8-B)

Corrosion effect is included by reducing the plate thickness by 1/16 inch (3.1.1.5). Is it possible that liner corrosion could introduce local flaws that would not reduce overall stiffness but introduce significant stress concentration points?

Response:

For propagation of a local flaw see the response to Question 220.40(b).

The corrosion allowance as described in PSAR Appendix 3.8-B, paragraph 3.1.1.5, is based upon the guidelines established in ASME B&PV Code, Section VIII, Division 1 for components exposed to water or steam. The cell liner system under normal operating conditions is exposed to an inert atmosphere on one side, and is coated with inorganic zinc paint on both sides thus inhibiting corrosion. On the back face of the liner plate, the liner is in contact with the 1/4 inch air gap at liner wall/ceiling locations and 1/8 inch air gap below the liner floor plate. As noted in the response to Question 130.95(1) the potential amount of moisture within the air gap is limited. Accordingly the consideration of the 1/16 inch corrossion allowance is conservative.

The possibility of a localized zone of behind the liner corrosion cannot be precluded. However, due to the conservative assumption of a corrosion allowance based upon ASME B&PV Code, Section VIII Division 1 requirements for steam and water vessels, coupled with the use of inorganic zinc paint on all surfaces of the liner plate to prevent corrosion, the anticipated corrosion flaw would have a thickness less than the corrosion allowance.

Question CS 220.36 (b) (Appendix 3.8-B)

Handling the corrosion by reducing plate thickness gives a lower stiffness. However, could the stiffness reduction erroneously give thermal stresses that are too low? Generally, the more flexible the structure, the lower the thermal stresses.

Response

In a fully restrained flat plate, the strains due to a uniform temperature are independent of the plate thickness. Because of local buckling, the liner strains differ from those of a fully restrained flat plate. An analysis with a mathematical model shown in Figure 3A.8-3 in the PSAR was performed using a reduced plate thickness (5/16 inch). The resultant strains are not significantly different from those shown in Figure 3A.8-3 for a 3/8 inch plate.

Question CS 220.36 (c) (Appendix 3.8-B)

Near very stiff areas on the liner boundary, for instance close to the pipe penetrations, the neighboring structure will have to exhibit considerable "give" or the liner attachment to anchors could be over stressed (Reference Figure 3A.8-8 where a steel anchor is apparently within 3 in. of the penetration collar). This will occur because the liner plant cannot buckle for short, unsupported spans. At what location in the structure will this situation be the worst and what are the shearing forces and displacements at the studs?

Response

Figure 3A.8-8 incorrectly depicted a stud anchor within 3 in. of the pipe penetration collar. Figure 3A.8-8 has been revised to reflect the configuration used in the analysis and specified for the CRBRP cell liner anchorage.

The liner stud anchor nearest to the penetration collar and/or embedment plate is located a minimum of 10 inches from the interface of the liner and the penetration/embedment.

The worst location for studs and plate is in the vicinity of "hard" spots such as embedments or penetrations. This has been covered in the reply to questions 220.36(d) and 220.40(a) which discuss the anchor and liner shear forces and displacements.

Question CS 220.36 (d) (Appendix 3.8-B)

At the top of page 3A.8-6 an analysis is described in which the panel corners at the stud anchors are assumed to be rigidly supported. This requires no unbalanced lateral forces on the anchors. If all panel sections buckle in the same direction this assumption is good. However, especially in the case of flat liner plates, the most probable buckling pattern may involve a shape where adjacent panels alternately buckle in and out. Has this case been analyzed and are the resulting shear loads in the panel at anchor attachment points acceptable?

Response

Analyses in which liner plates buckled in and out were performed. Figures 220.36(d)-1 and 220.36(d)-2 show two of the cases, which are in the vicinity of a penetration and an embedment respectively. The liner criteria is defined in terms of allowable equivalent von Mises strains and the calculated values (shown in Figures Q220.36(d)-1 through 4) are within the allowable limits. The principal shear stresses in the plate and in the stud were calculated and are well below one half of the ultimate strength of the material.

To prevent liner tear by the stud, the ratio of stud diameter over plate thickness (d/t) was limited conservatively to a value of 2.0. This is based on Reference QCS 220.36(d)-1 which reports the results of testing of plate stud systems. Reference QCS 220.36(d)-1 reports that plate failure occurs at a plate thickness to stud diameter ratio (d/t) of 2.7 or greater.

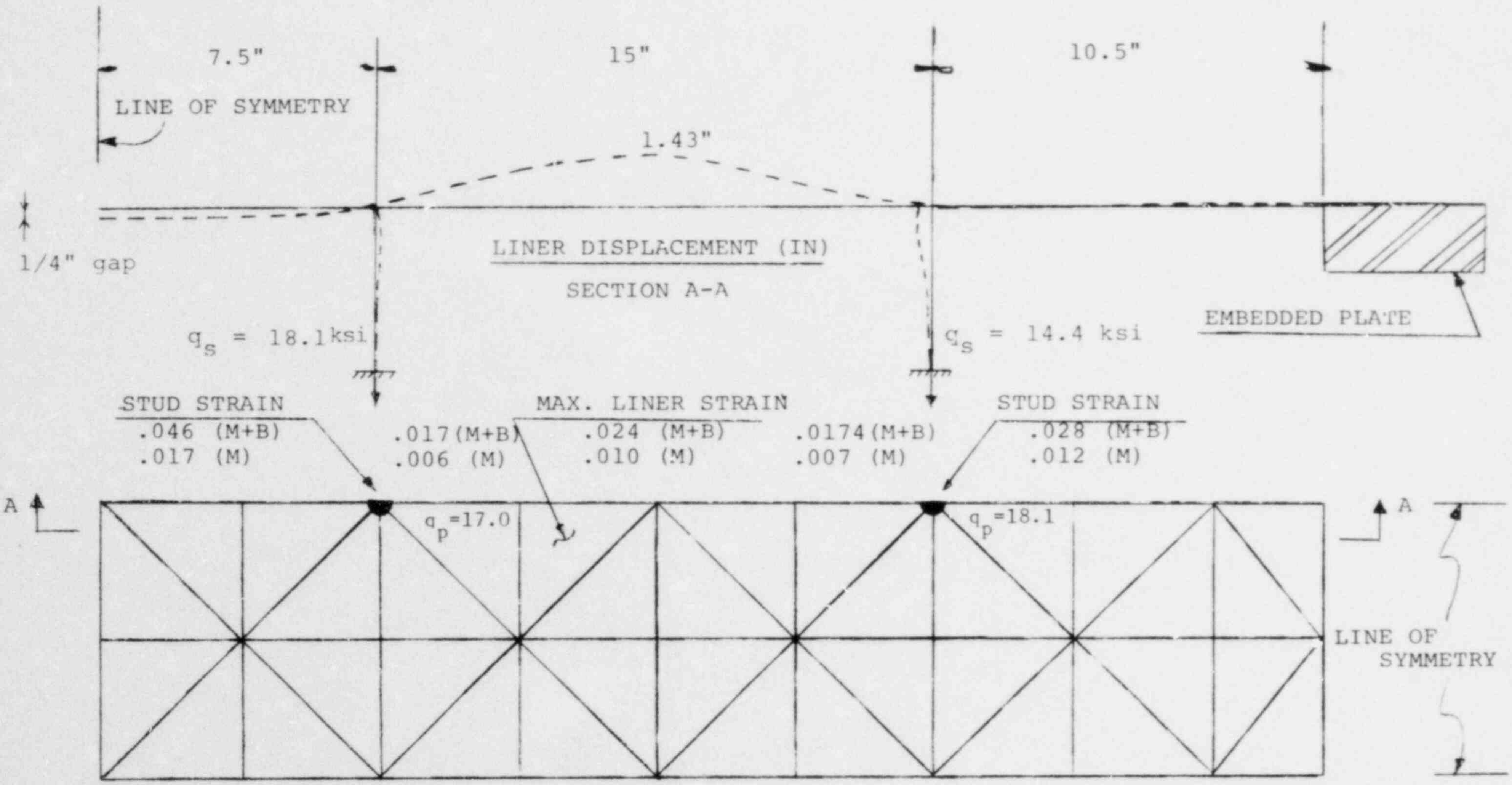
Reference

Q220.36(d)-1 G. G. Goble, "Shear Strength of Thin Flange Composite Specimens," AISC Engineering Journal, April 1968

ULTIMATE STRENGTH AND STRESS AT TEMPERATURE

q_s = Principal Shear Stress in Stud (Ksi)

q_p = Principal Shear Stress in Liner (Ksi); Maximum at a Stud Location



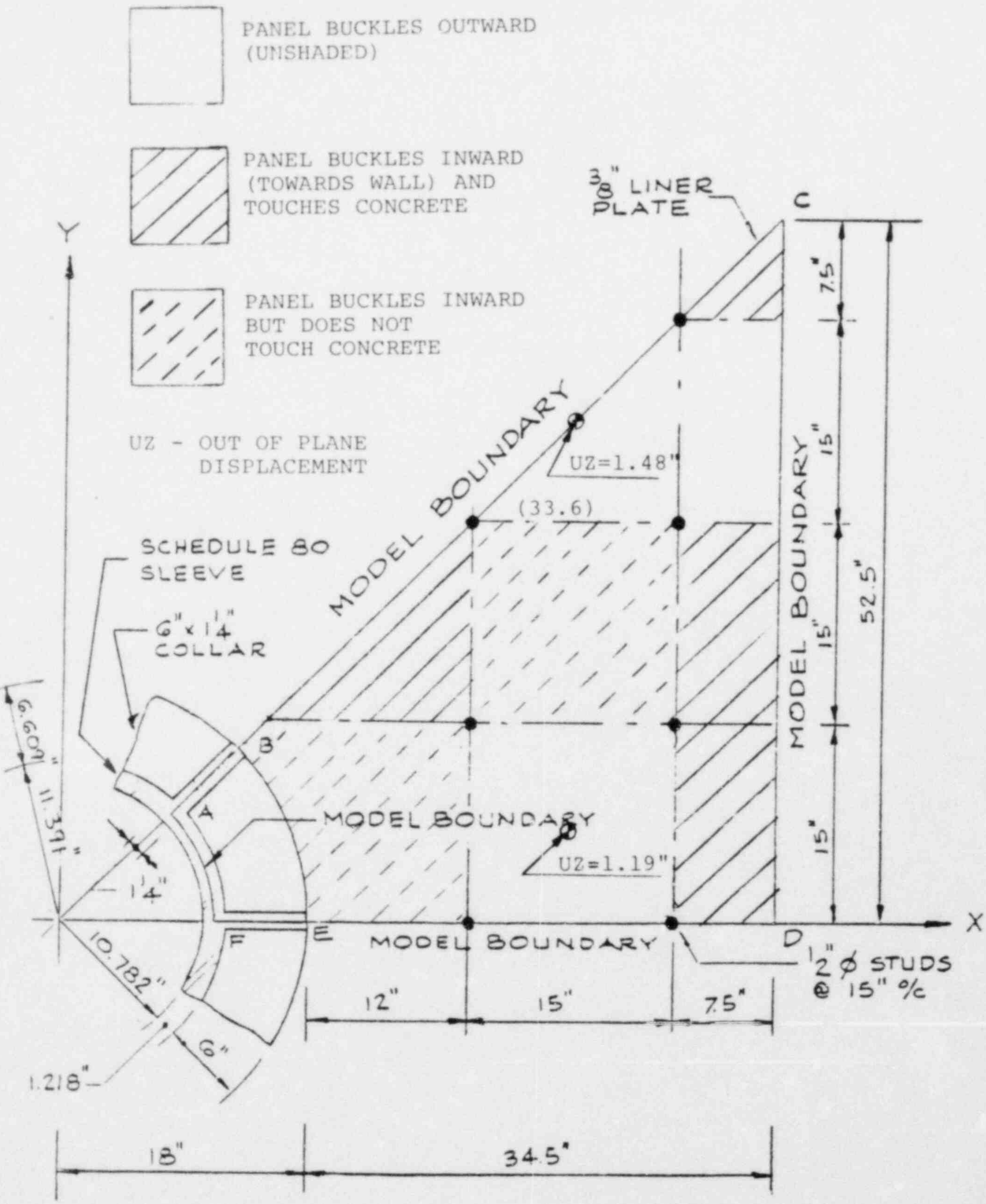
M = Membrane Strain

M + B = Membrane plus Bending Strain

MAXIMUM EQUIVALENT STRAINS (IN/IN)

WALL LINER AT EMBEDDED PLATES, BUCKLING PATTERN

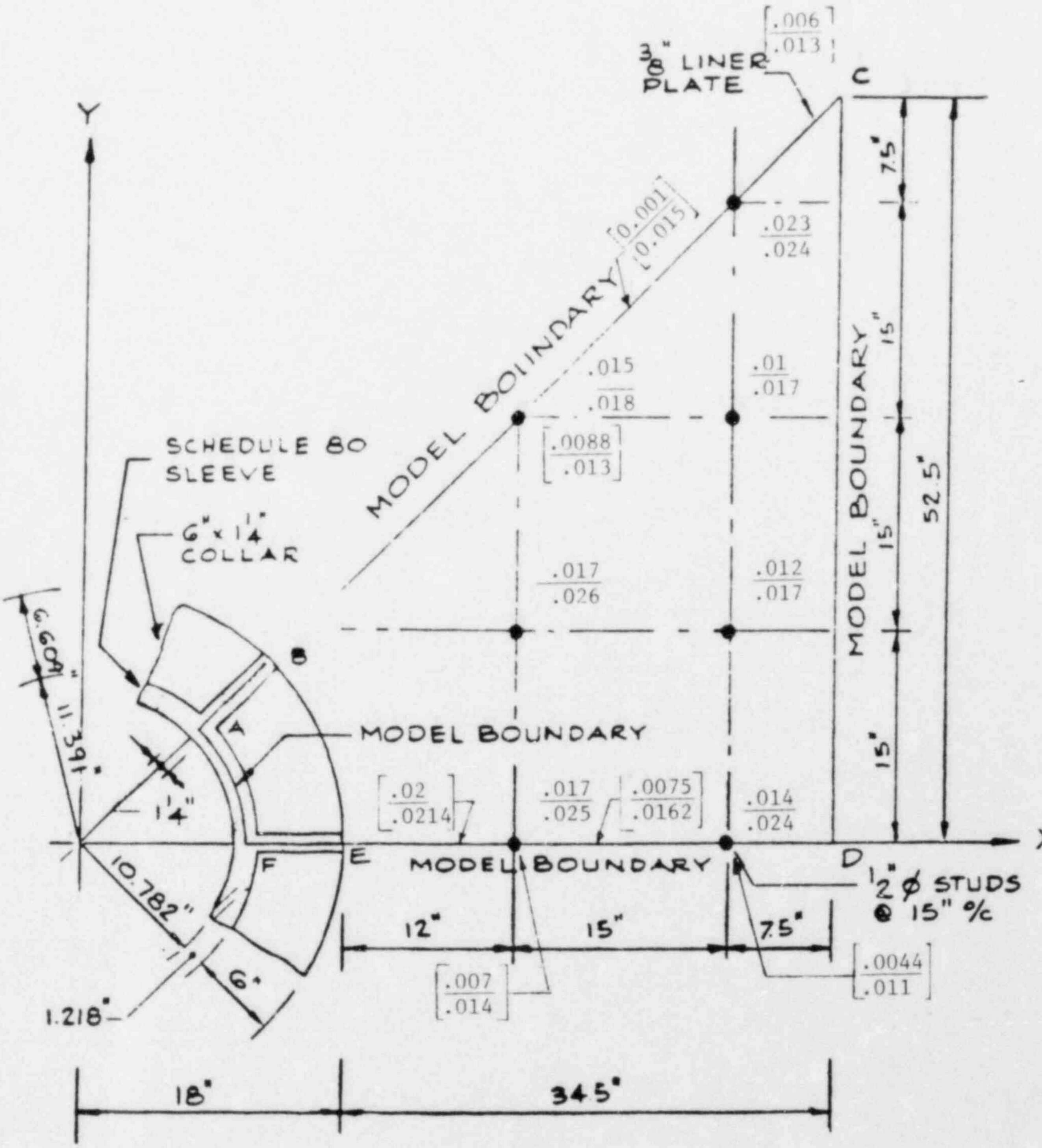
FIGURE Q220.36(d)-1



WALL LINER AT PENETRATION; BUCKLING PATTERN
FIGURE Q220.36(d)-2

LINER STRAIN (IN/IN) $\left[\frac{\text{MEMBRANE}}{\text{MEMBRANE + BENDING}}\right]$

STUD STRAIN (IN/IN) $\frac{\text{MEMBRANE}}{\text{MEMBRANE + BENDING}}$



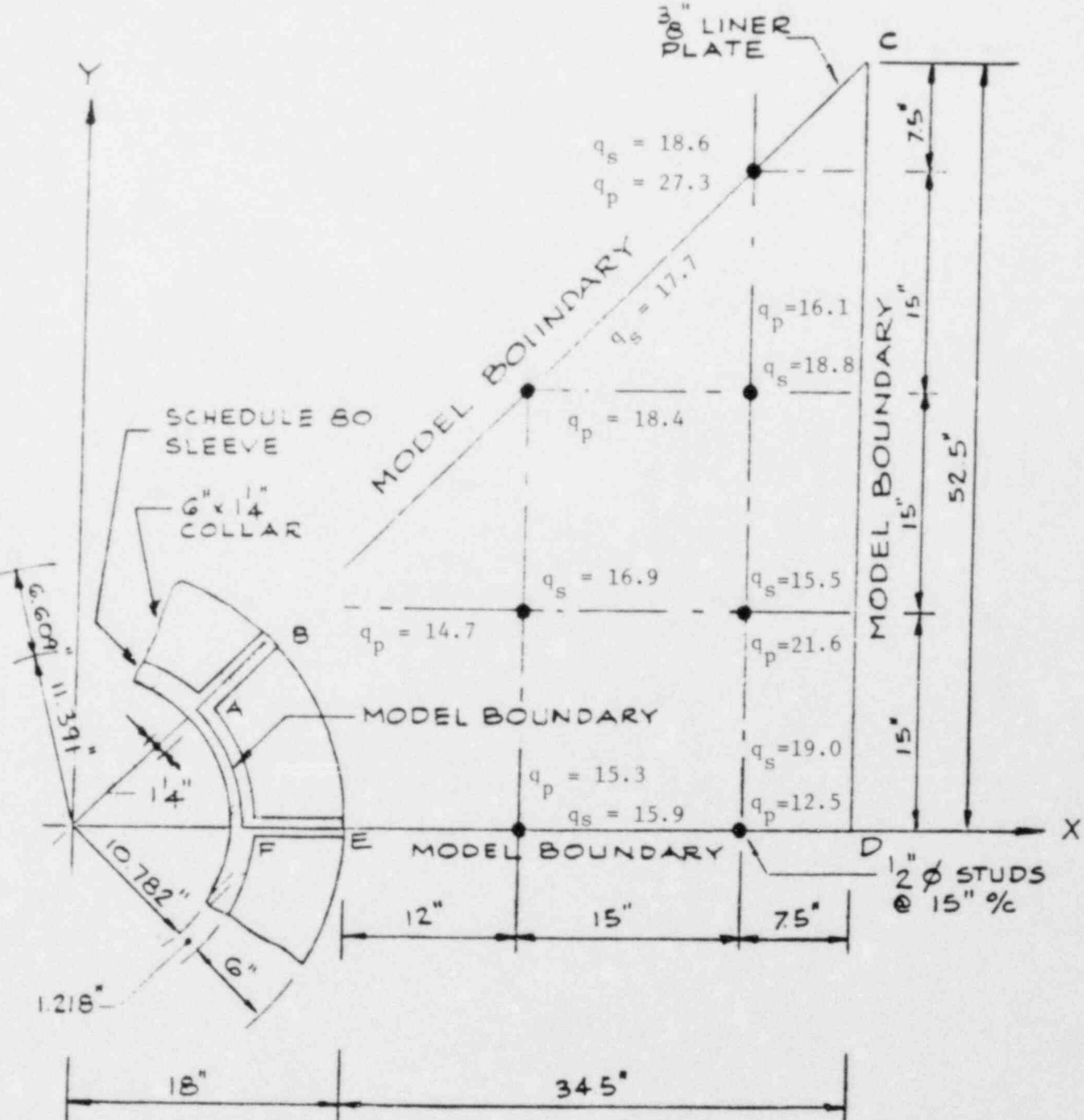
WALL LINER AT PENETRATIONS; GENERALIZED STRAINS

FIGURE Q220.36(d)-3

ULTIMATE STRENGTH AND STRESS AT TEMPERATURE

q_s = Principal Shear in Stud (Ksi)

q_p = Principal Shear in Liner (Ksi); Maximum at a Stud Location



WALL LINER AT PENETRATIONS; SHEAR STRESSES IN LINER AND STUDS

FIGURE Q220.36(d)-4

Question CS 220.36 (e) (Appendix 3.8-B)

In all analyses presented, uniform temperature distributions over the liners are assumed. Are these areas where this assumption is not valid? If so, are the stresses generated acceptable? Unequal thermal expansion on either side of a stud could generate considerable lateral force on the stud.

Response

The liner system has been analyzed assuming a uniform temperature distribution. Localized hot spots resulting from sodium jets or localized sodium spills and the boundary between the wetted and non-wetted areas of the walls will be subjected to non-uniform temperatures. These events could generate lateral forces on the anchors, when the hot plate tends to expand. However, the liner plate in the colder areas provides restraint to thermal growth and will limit the possible lateral displacements of the anchors. The response to Question CS 220.36(d) describes conditions in the vicinity of "hard" spots (penetration or embedment) where the thermal expansion of a thicker plate imposes lateral displacements to the liner studs and shows that the studs and liner are capable to withstand those conditions. A hot spot presents a similar situation, but the colder liner plate around the hot spot will provide more restraint than the hot plate around the "hard" spot. Therefore the stress/strain conditions of the stud and liner will be less severe.

Question CS 220.36 (f) (Appendix 3.8-B)

When the liner buckles and bears against the insulation concrete, considerable tensile loads are generated in the studs. If the studs don't give, the liner could fail in a shearing mode at the stud connection. Has this possibility been evaluated? If so, what are the results?

Response

The response to this question is included in the response to Question CS 220.40(a).

Question CS 220.37 (a) (CRBRP-3, Vol. 2)

Table 3-14 gives results for a submerged liner without creep. The applicant should quantify how much actual strains are reduced when creep is taken into account. Does the applicant really mean creep, or is stress relaxation a more appropriate term? Under conditions of creep, will the ultimate strain capability of the liner material change significantly?

Response

The term "creep" was used in Table 3-14 to refer to the "time effects" on the load deformation characteristics of the material. It is agreed that the term "stress relaxation" would have been more appropriate for a liner subjected to temperature effects.

An analysis that included the "time effect" on the load deformation characteristics was done using a model of a restrained panel anchored to the concrete with a stud. This analysis was done for qualitative rather than quantirative purposes and indicated a general relief of stresses and strains. The analysis was carried only up to 1600°F and a total time of 12 hours, however, due to the qualitative nature of the analysis certain simplifications were made in assessing the time effects. Thus, instead of applying the time effect continuously it was lumped at certain selected levels of temperature. The results indicated that above T = 1200°F there was sufficient relaxation to prevent further plastic deformations.

The time etfects on the load deformation characteristics result in relaxation of stresses and strains will have no significant effect on the ultimate strain capability of the material.

Question CS 220.37 (b) (CRBRP-3, Vol. 2)

In Section C.3.4.4, the applicant proposes to use "von Mises effective strain" for the liner failure criterion. Keeping in mind that the strain is not necessarily linearly related to stress beyond yield, a rigorous definition of what "von Mises strain" means beyond the yield point is needed.

Ultimate strength is often used to predict failure when the facture mode is known to be simple cohesive failure. For general ductile fracture, especially when shear fracture is a strong possibility, maximum shear stress is a preferred failure criterion. The liner can be expected to develop considerable shear stresses, especially near the anchor studs.

Considering the above comments, what is the justification for using "von Mises strain" as the failure criterion? Should maximum shear stresses also be considered?

Response

The relations that describe elastic plastic behavior, based on the Prandtl-Reuss flow rule, are expressed in terms of an equivalent or effective stress, σ_e, and an equivalent or effective plastic strain increment de_o (Ref. QCS220.37(b)-1). The expression for the equivalent stress has the same form as the von Mises yield criterion, σ_e, so that just as yielding begins the two are equal. An expression for the equivalent or effective strain ε_e also has the same form as the strain ε_e corresponding to the von Mises yield criterion σ_e. Beyond yield the relationship between the effective stress and the equivalent or effective strain is taken from the uniaxial tensile stress strain curve as explained in Ref. Q220.37(b)-1.

Since the expression for the equivalent or effective strain has the same form as the strain corresponding to the von Mises yield stress criterion the term generalized von Mises strain and equivalent or effective strain or simply generalized strain have been used in CRBRP documents interchangeably. The effective stress and effective strain are defined by the following equations:

$$\sigma_e = \frac{1}{\sqrt{2}}\left[(\sigma_1-\sigma_2)^2+(\sigma_2-\sigma_3)^2+(\sigma_3-\sigma_1)^2\right]^{1/2}$$

$$\varepsilon_e = \frac{\sqrt{2}}{3}\left[(\varepsilon_1-\varepsilon_2)^2+(\varepsilon_2-\varepsilon_3)^2+(\varepsilon_3-\varepsilon_1)^2\right]^{1/2}$$

Where:

$\sigma_1, \sigma_2, \sigma_3$ are the principal stresses

$\varepsilon_1, \varepsilon_2, \varepsilon_3$ are the total principal strains

It should be pointed out that Section III, Division 1 of the ASME B&PV Code accepts the yield criterion and associated flow rule based on the energy of distortion method (von Mises), for plastic analysis of nuclear power plant components (Appendix F, Section F-1321.1.c).

The above subject is also discussed in Appendix 3.8-B, Attachment D of the PSAR (page 3.8-B.18), which justifies the use of von Mises strain as the failure criterion.

The 3/8" carbon steel liner plate used in CRBRP is a very flexible structural element particularly at elevated temperatures where there is a significant reduction of the stiffness. Due to its flexibility the liner can undergo large deformations without rupture and its behavior is similar to that of a flexible membrane. Thus, shear forces from the stud anchor are resisted primarily by membrane forces that develop as the liner stretches to large deformations. For this reason shear fracture is not considered a possibility. Further discussion on this subject is provided in the response to question 220.36.

Reference

QCS 220.37(b)-1 Mendelson, A. Plasticity: Theory and Application, The MacMillan Company, New York, 1968.

Question CS 220.38

During the TMBDB accident scenario several modes of failure for concrete internal structures are considered. One of these is termed "section failure" and occurs when the moment capacity of a section is exceeded (for example, the floor of a pipeway cell). This failure mode involves large displacements (rotations) and therefore could result in failing the cell liner allowing additional sodium-concrete interaction.

The applicant has provided information showing that such section failure do not occur before TMBDB requirements are met. Additional details should be provided to show how close these internal structures come to failure as the TMBDB scenario progresses. Rates at which failure is approached and times that failure is expected should also be provided. This should be done for all critical sections. This is particularly important because no factor of safety is applied to the section failure capacities.

Response

Critical sections in the RCB structures have been or are being evaluated for rates of failure progression and expected failure times.

In accordance with the TMBDB scenario described in CRBRP 3, Vol. 2 the Reactor Cavity and the three Pipeway Cells are the only cells in the Reactor Containment Building that would be exposed to sodium. The base case scenario considers failure of the liner to occur at certain times and these failure times establish the minimum structural integrity requirements for the liner system. Since the liner system is anchored to concrete structures, failure of these structures (sectional capacity exceeded) is considered, conservatively, to imply potential liner failure so that the minimum times for structural integrity requirements for the liner apply also to the supporting concrete.

The reactor cavity and the pipeway cell structures were evaluated and found to meet the scenario requirements with substantial margin in most cases. A plot of the time variation of thermal moment and moment capacity is given in Figure Q220.38-1 for the upper portion of the reactor cavity wall. This plot provides information on how the thermal moment approaches the moment capacity for this particular section of the structure. Although the margin may not necessarily be as high for other parts of the cells in question the plot demonstrates that the capacity is not approached asymptotically in the time frame in question and the evaluations that demonstrate integrity, at the required times, provide sufficient assurance that scenario requirements are met.

The question makes specific reference to the pipeway cell floor and for this reason a brief discussion is given for this structure. The pipeway floor consist of two layers of concrete; a bottom layer which is anchored to the reactor cavity and the surrounding vertical walls, and a top layer which is free to expand. Th floor liner is anchored to the top layer and for this reason its integrity is not affected by the deformations of the bottom concrete layer as long as there is no collapse. In any case, liner integrity is required for only 30 hours and the capacity of the bottom concrete layer is not exceeded before the boil dry time (132 hours), so there is substantial margin.

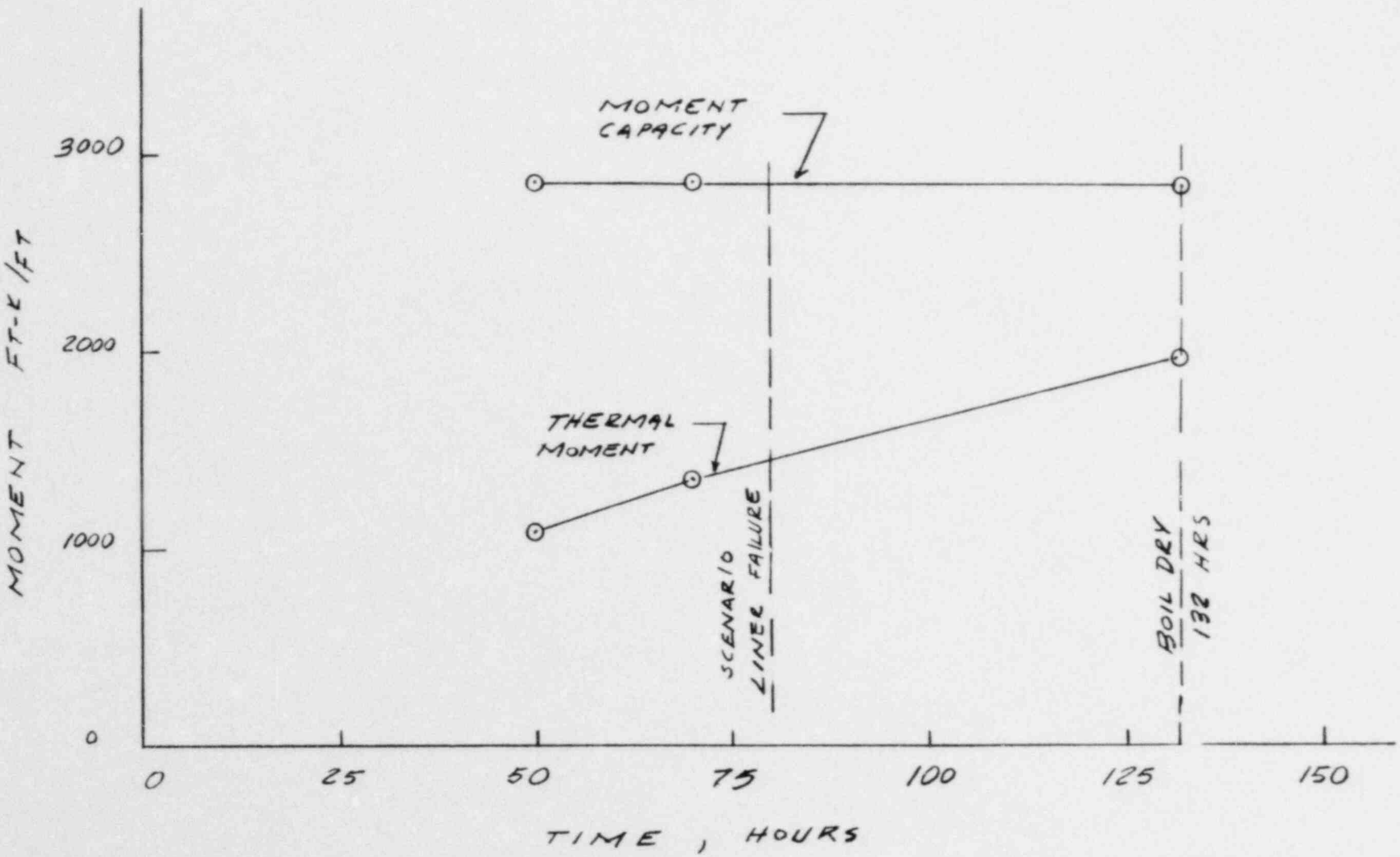


FIGURE Q220.38-1 REACTOR CAVITY WALL - UPPER REGION
TIME HISTORY OF THERMAL MOMENT AND MOMENT CAPACITY

Question CS 220.39

Information detailing the current version of the MPHI computer code must be provided along with benchmark information for validation of the code and its use in this particular application.

Response

MPHI is a computer program that calculates moment-curvature (M-ϕ) relationships for reinforced concrete sections under thermal gradients, and with an axial force P. The axial force P may be zero. The program has capabilities to account for non-linear material properties, variation of material properties with temperature, non-linear thermal gradients, tensile cracking, and compressive crushing. The M-ϕ relationship provides information on the capacity of a section under a temperature gradient. The moment corresponding to zero curvature is also of interest because it represents the thermal moment for a section restrained against rotation.

For a given axial force P and temperature distribution, the moment-curvature relationship is determined using a numerical procedure that involves the following:

1. The section under consideration (Figure Q220.39-1) is divided into a number of elements by nodal points. The thermal strain ε_t is calculated at each node, i, as:

   $(\varepsilon_t)_i = (T_i - T_{ref})\,\alpha_{(T\ )}$

   where

   T_i is the nodal temperature

   T_{ref} is the reference temperature

   $\alpha_{(T\ )}$ is the average coefficient of thermal expansion at T_i

2. A plane section is passed at a curvature ϕ with strains ε_a and ε_b at the two edges of the section. At node i the strain ε_i is:

   $$\varepsilon_i = \left(\frac{\varepsilon_a - \varepsilon_b}{H}\right)h_i + \varepsilon_b$$

   where

   H is the total depth of the section

   h_i is the coordinate of node i

   The mechanical strain at node i is:

   $$(\varepsilon_s)_i = \varepsilon_i - (\varepsilon_t)_i$$

3. Stresses, σ_i, are calculated at each node based on (ϵ_s) i and a stress-strain relationship which is defined in the input of the problem. Element forces are calculated based on the average of the stresses in the two nodes defining the element.

4. The axial force, P', and the moment, M, are calculated by summation of the element forces and their moments about the centroidal axis.

5. The value of P' is compared with the force under consideration (P), and if different the plane section is moved to a new position maintaining the same curvature ϕ and the process is repeated until convergence.

6. A new curvature is selected and steps 2 to 5 are repeated.

7. In this manner a ϕ versus M relationship is developed for a given P.

The program has options to develop the moment-curvature relationship for the following axial force, P, or restraint conditions.

a. Axial force specified in the input.
b. No axial restraint (P = 0, displ. $\neq$ 0)
c. Full axial restraint (P $\neq$ 0, displ. = 0)

The last two options are special cases of the first and provide the capability to develop mement curvature relationships for two extreme cases of axial restraint. The first option may be used for specific axial force values.

Tensile cracking of the concrete is accounted for automatically by the imput σ-ϵ relations. Compressive crushing of a concrete element is assumed to occur in the part of the section where the strains exceed a limiting value which is input to the program. Degradation of a concrete element is assumed to occur in the part of the section where the temperature of the element exceeds a limiting value specified in the program. The part of the section where the strain or the temperature exceeds the limiting value is automatically removed from the section.

Availability

The MPHI program was developed by Burns and Roe and is available as a Burns and Roe in-house program in the Burns and Roe computer and in time sharing CDC computers.

Verification

The program was verified by hand calculations. For this purpose a reinforced concrete section was considered under a temperature distribution and was divided into elements (Figure Q220.39-2). Material properties, for the purpose of this calculation, were assumed to be those shown in Figures Q220.39-3 to Q220.39-5. The M-ϕ relationship was developed for the case of full axial restraint and then, selected points on the relationship were calculated by hand. The moment curvature points obtained by hand calculations are in complete agreement as shown in Table Q220.39-1. It should be pointed out that in addition to the values in Table Q220.39-1 the hand calculations provided a detailed check for the intermediate steps of the computer program such as strains and stresses to ensure that there are no errors that might affect the results under a different set of variables.

Further verification of the program was performed using the computer program ANSYS (Reference Q220.39-1). Limited analysis by ANSYS provided information for the moment corresponding to zero curvature (thermal moment) for the section in Figure Q220.39-2 under full axial and no axial restraint. The model is shown in Figure Q220.39-6 and the properties in Figures Q220.39-3 to Q220.39-5. A comparison of the MPHI and ANSYS results is given in Table Q220.39-2.

Application

MPHI is used to calculate the moment-curvature relationship of reinforced concrete sections under temperature distribution and axial force. The moment capacity may be obtained from the M-ϕ relationship. In addition the thermal moment of a section restrained against rotation may be obtained as that corresponding to zero curvature. (Figure CSQ220.39-7)

Reference

QCS 220.39-1 Computer Program ANSYS, Revision 3, Swanson Analysis Systems, Inc., Houston Pennsylvania.

TABLE QCS220.39-1

MPHI VERIFICATION RESULTS

Curvature 1/in.	Computer Results Force lbs	Computer Results Moment lbs-in	Hand Calculations Force lbs	Hand Calculations Moment lbs-in
0	-53,725	-27,327	-53,725	-27,327
+.00002	-53,892	+27,268	-53,892	+27,268
-.00002	-52,879	-76,738	-52,879	-76,738

Moment is positive when it creates tension on top (Node No. 1).
Negative axial force causes compression on the section.

TABLE QCS220.39-2

COMPARISON OF RESULTS FROM MPHI AND ANSYS

CASE	MPHI RESULTS FORCE KIPS	MPHI RESULTS MOMENT k-in	ANSYS RESULTS FORCE KIPS	ANSYS RESULTS MOMENT k-in
Curvature $\phi = 0$				
Full Axial Restraint	-53.7	-27	-53.2	-31
No Axial Restraint	0	-193	0	-188

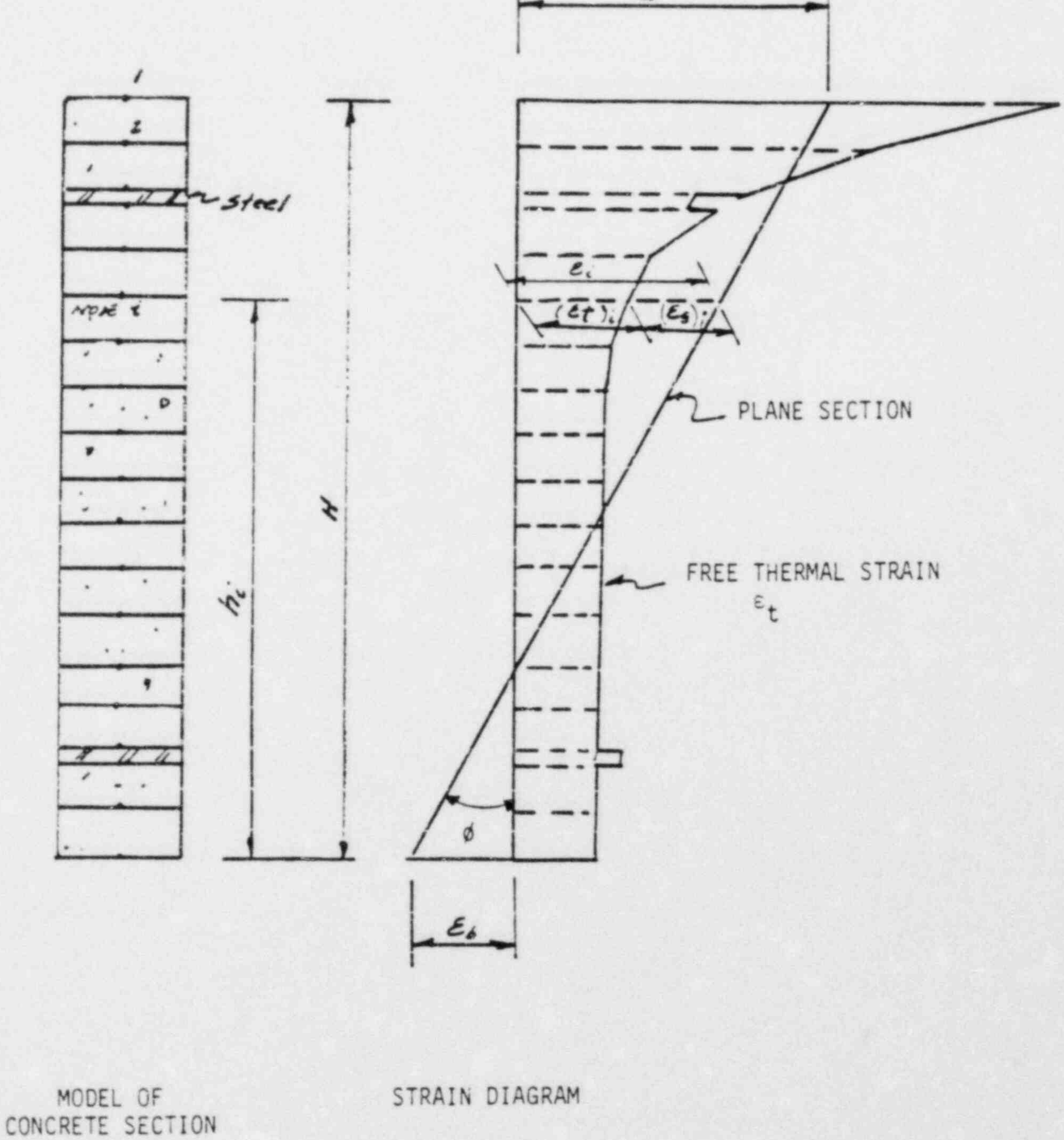


FIGURE Q220.39-1 TYPICAL MPHI MODEL AND STRAIN DIAGRAM

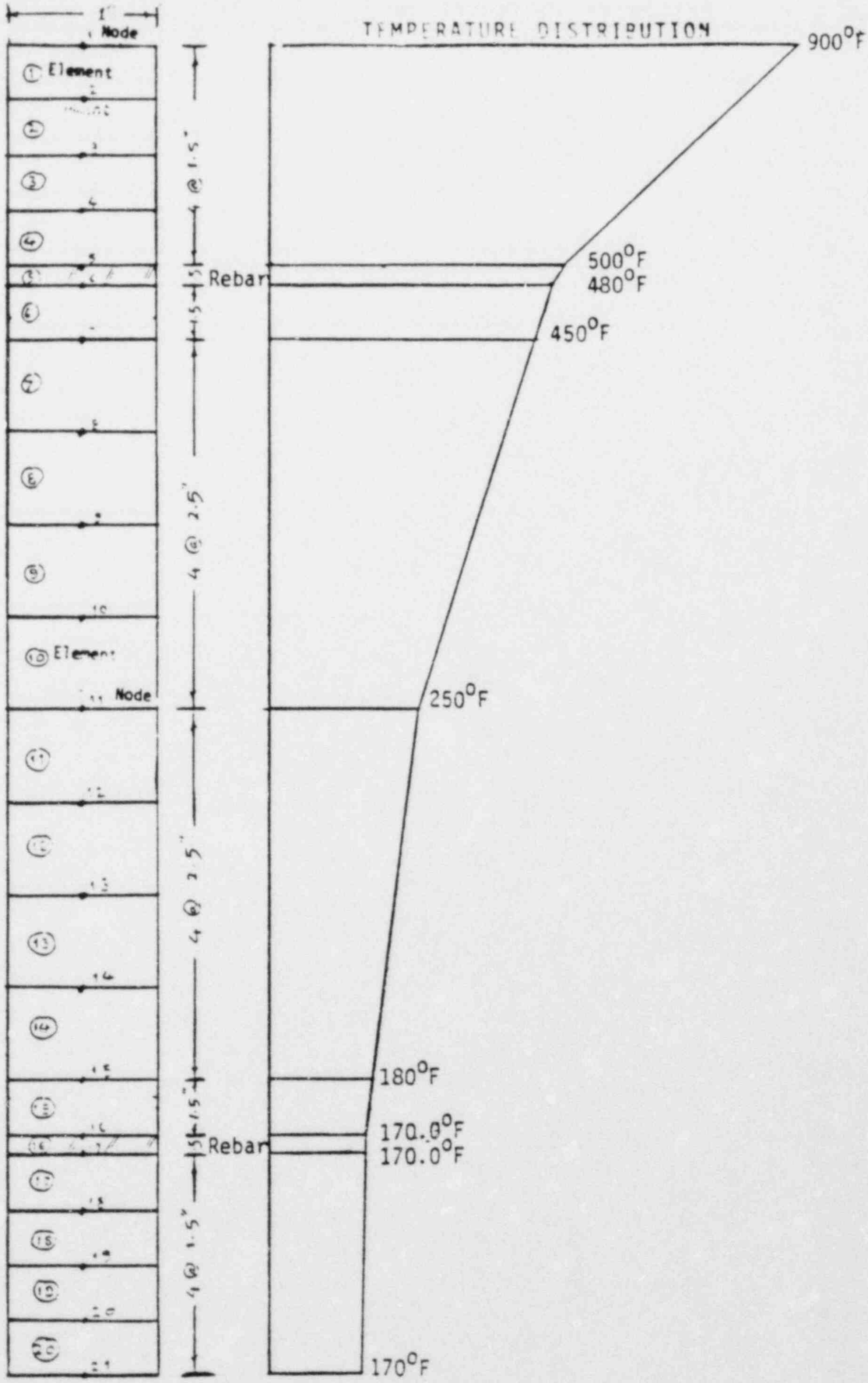


FIGURE Q220.39-2 MODEL AND TEMPERATURES FOR VERIFICATION PROBLEM

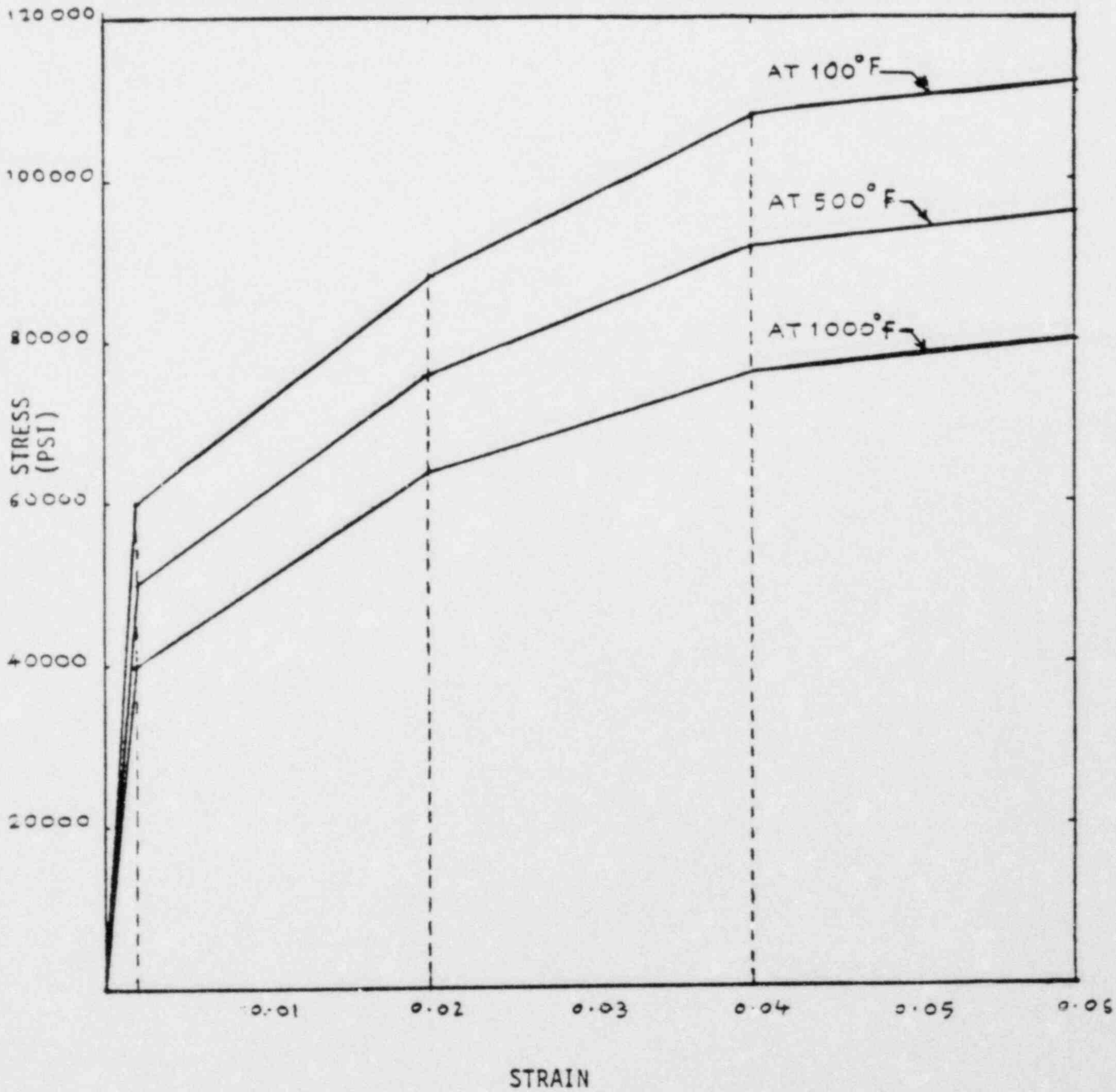


FIGURE Q220.39-3 STRESS-STRAIN CURVES FOR REINFORCING BARS VERIFICATION PROBLEM

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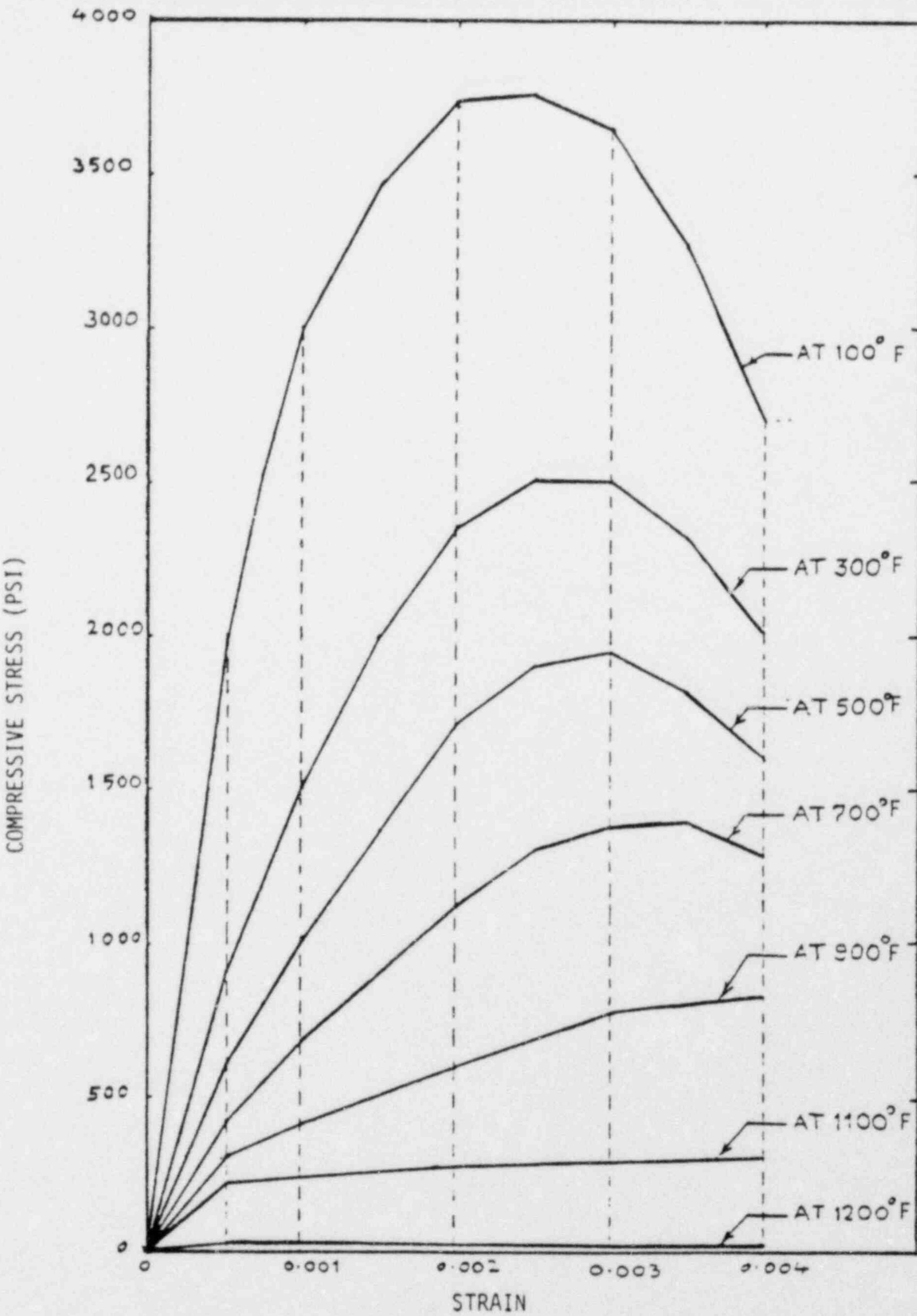


NOTE: Stresses in concrete are zero for tensile strains

FIGURE Q220.39-4 STRESS-STRAIN CURVES FOR CONCRETE VERIFICATION PROBLEM

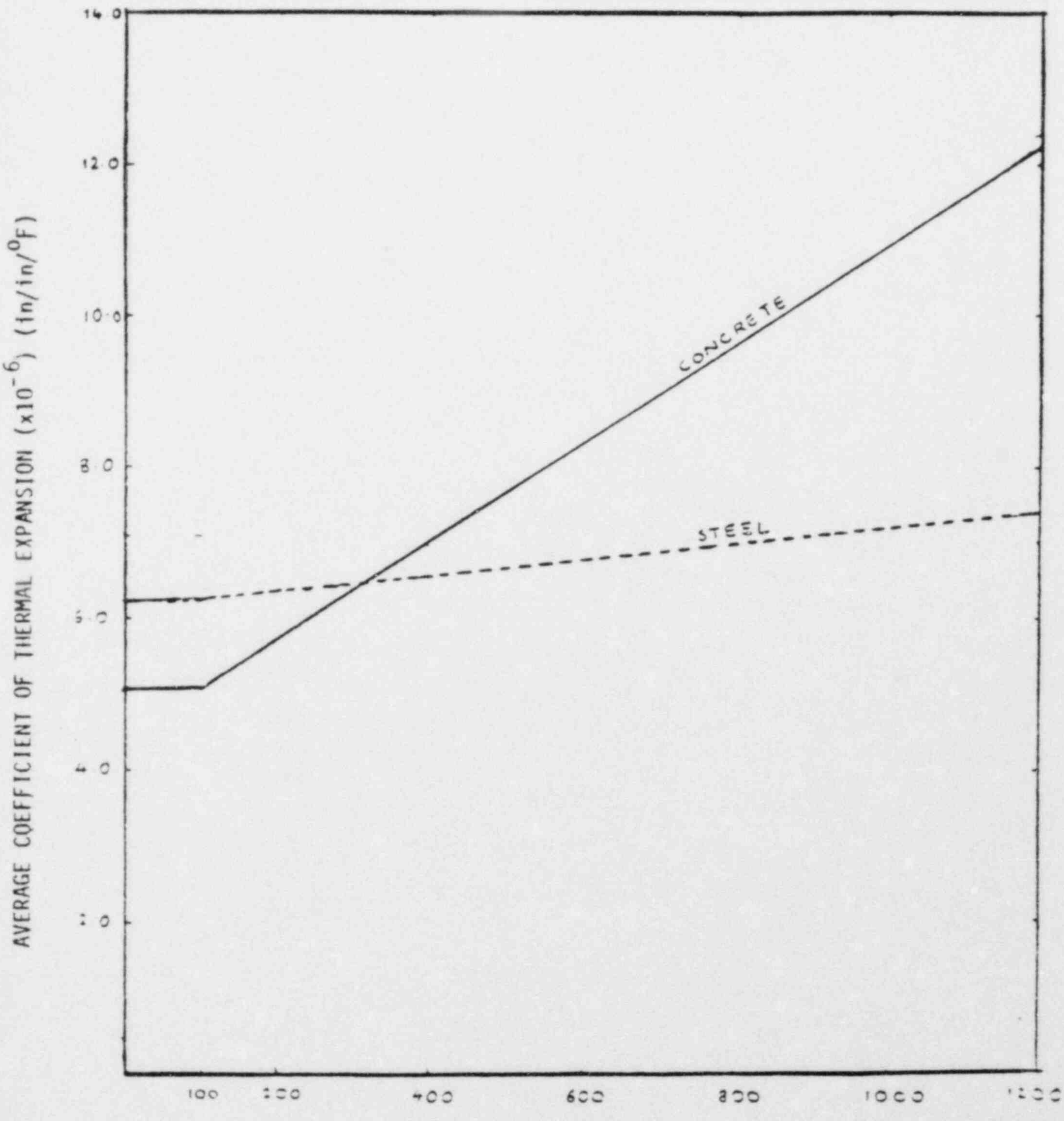


FIGURE Q220.39-5 COEFFICIENT OF THERMAL EXPANSION - VERIFICATION PROBLEM

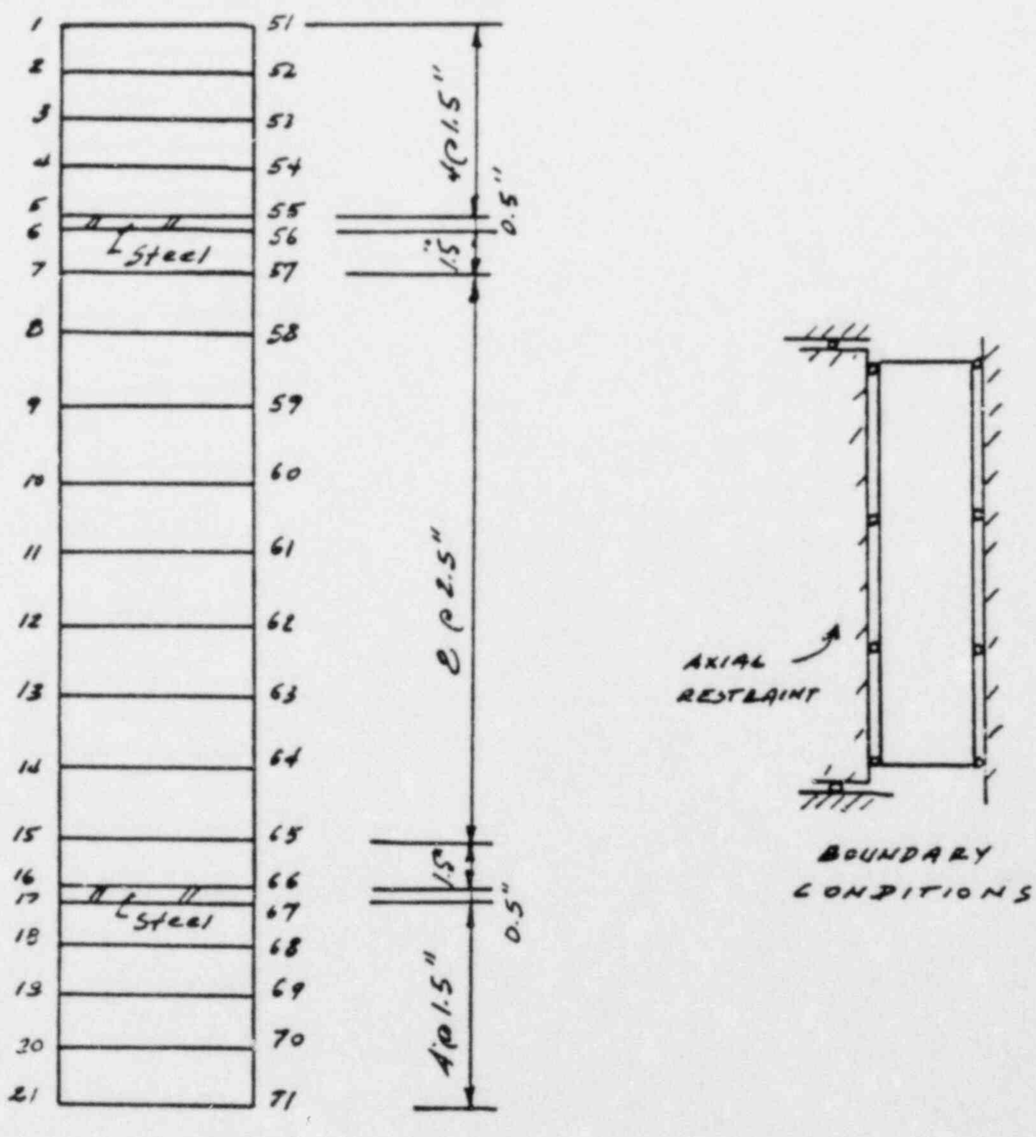


FIGURE Q220.39-6 ANSYS MODEL OF REINFORCED CONCRETE - ϕ = 0

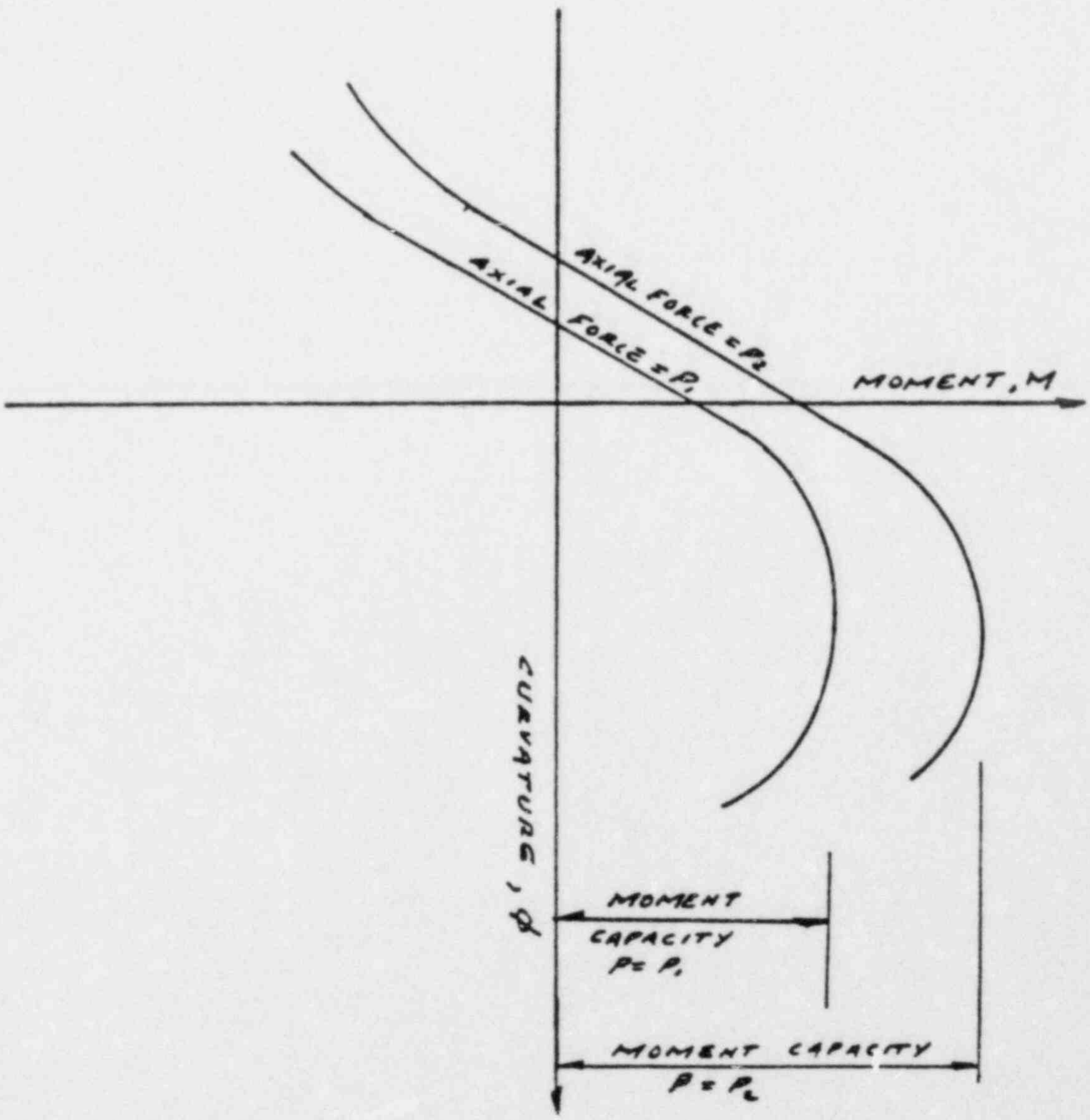


FIGURE Q220.39-7 TYPICAL MOMENT-CURVATURE DIAGRAMS

Question CS 220.40 (a)

The applicant needs to determine the most likely buckling mode for the cell liner. If the shear symmetric mode is expected, local stresses near stud anchors must be evaluated to determine shear stress state and provided in the PSAR.

Response

The cell liner analysis performed indicate that the most likely buckling pattern for the cell liner is that of alternate panels buckling in and out in a checkerboard pattern. This is discussed in the reply to Question CS 220.36(d). The reply to this question also describes the criteria adopted to prevent plate/stud tear out failures.

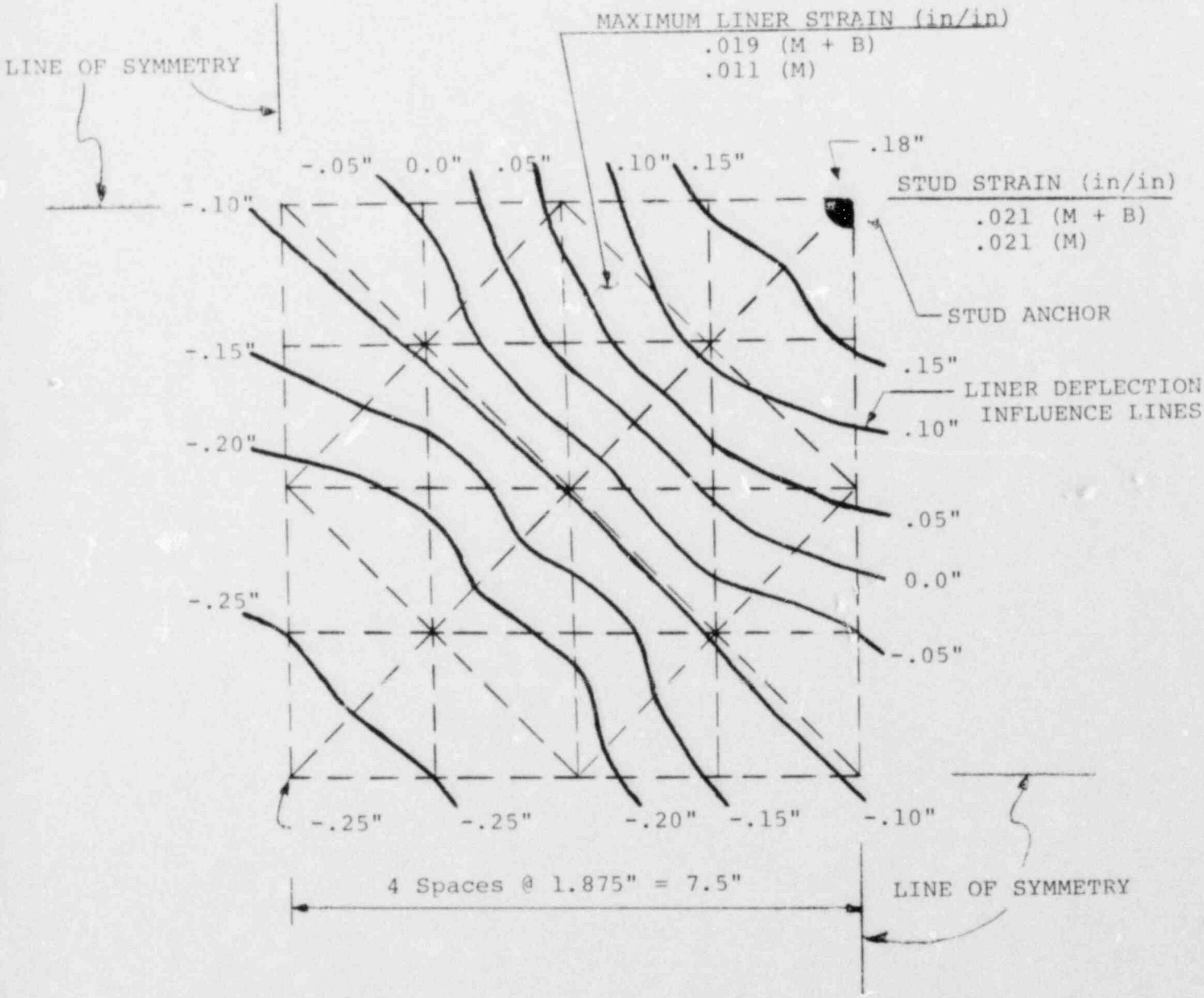
A case of symmetric buckling, where the liner bears against the concrete and imposes a large tensile load on the stud anchor, has been examined. See Figures QCS220.40(a)-1 and QCS220.40(a)-2. As a result of large deformations associated with the plastic yielding of the liner plate and stud, membrane compression develops in the liner in equilibrium with the stud tension. Consequently, no significant shear stresses can be developed in the liner. Strain levels in the liner and stud under this mode are within the acceptable limits.

In a further evaluation of shear effects, it was assumed that there is no membrane action and that punching shear develops in the plate. The shear stresses were calculated based on the ultimate strength capacity of the stud. The shear stresses in the plate were determined to be respectively 35% and 48% of the ultimate tensile strength of the plate for 1/2 inch and 3/4 inch diameter studs.

MAXIMUM GENERALIZED STRAINS

M = MEMBRANE STRAIN

M + B = MEMBRANE PLUS BENDING STRAIN



CELL LINER - SYMMETRIC BUCKLING; TENSION IN STUD

FIGURE Q220.40(a)-1

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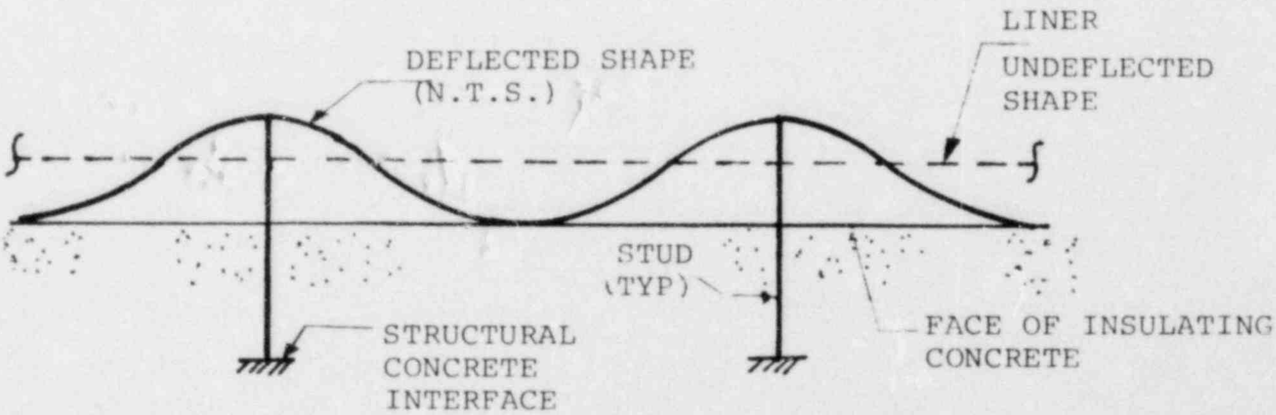

CELL LINER - DEFLECTED SHAPE - SYMMETRIC BUCKLING

FIGURE Q220.40(a)-2

Question CS 220.40 (b)

Evaluate the effect of pre-existing cracks and of cracks generated during the life of the plant; discuss possible propagation of these cracks in the liner.

Response

The cyclic loading on the cell liner is such that in accordance with ASME B&PV code Section III, Division 1, Section NE-3222.4 a fatigue analysis is not required. Therefore, it is not expected that existing flaws, if any, will propagate in the cell liner.

Additionally, the potential for cell liner brittle fracture dure to radiation has been investigated in accordance with the methods and limits established by USNRC Regulatory Guide 1.99 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials". In the worst case exposure condition, at the reactor cavity beltline, the resultant maximum adjustment in the Nil-Ductility Transition (NDT) temperature is 10°F. This indicates that the cell liner is not affected by neutron embrittlement.

The cell liner system is designed, fabricated, installed, and inspected as an Engineered Safety Feature. The requirements for the ESF cell liners are defined in PSAR Section 3.8-B.

The materials, fabrication, welding and construction NDE requirements identified in PSAR Appendix 3.8-B, Attachment B, preclude the incorporation of defects in the cell liner which are capable of propagation. The acceptance standards for cell liners have been extracted from ASME B&PV Code Section III, Division 2.

The generation of cracks in the cell liner system during the plant lifetime is prevented by the design. The strain limitations imposed for DBA conditions preclude plastic instability.

The propagation of cracks in the cell liner is prevented from impacting the response of the cell liner under DBA sodium spills by both scheduled and unscheduled inservice inspection requirements which are identified in PSAR Appendix 3.8-B, Attachment E. The scheduled inservice inspection program assures the continued integrity of the overall cell liner. Unscheduled inservice inspection is required in the event of a cell liner repair, small sodium spill, any indication of corrosion in excess of design limits, any thermal cycling in excess of the duty cycle or indications of excessive oxygen in-leakage.

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Question CS 220.40 (c)

In addition to the non-uniform temperature distribution in liner to be considered (Ref. Question 220.36e) the response to a shallow pool spill (for instance localized in a very stiff area such as a corner) should be evaluated.

Response

The response of the cell liner to a shallow sodium spill, localized in the region of a cell liner corner, has been evaluated. An analysis has been performed using the Wall-Floor Liner corner model shown in Figure 3A.8-3. The analysis assumes the cell liner floor panel, located adjacent to the corner, to be heated to 1000°F while the remainder of the cell liner is maintained at 70°F. The induced strains in the liner are lower than the strains reported in Section 3A.8.3.3 for the uniformly heated liner case.

Actual conditions preclude such a confined hot area since the conductivity of the plates will result in a rapid redistribution of heat.

Question CS 220.41

The portion of containment at cells that experience pressure over 10 psig may have to be qualified as ASME Section III, Division 2 concrete pressure vessels because the concrete is relied upon to carry a portion of the pressure load, unless applicant can justify his position in the PSAR.

Response

The RCB lined cells are not a part of the Containment System. As safety related reinforced concrete structures the cells are designed per ACI 349. This code includes design requirements for pressure and other loads on safety related concrete structures. Therefore the ASME Section III, Division 2 Code is not applicable in this case.

Question CS 220.42

The applicant has performed bounding seismic response calculations by using "soft" and "stiff" soil springs. The applicant is required to show that intermediate values of soil stiffness will not increase the response levels. This could be shown by comparing modal frequencies of major contributing modes to the response spectrum.

Response

The attached Figure QCS 220.42-1 compares the response spectra at the operating floor for the following soil properties:

a) upper bound

b) average

c) lower bound

The curve for the average properties falls within the band of upper and lower bound properties for the entire frequency range under consideration.

Thus, intermediate values of soil stiffness will not increase the response levels.

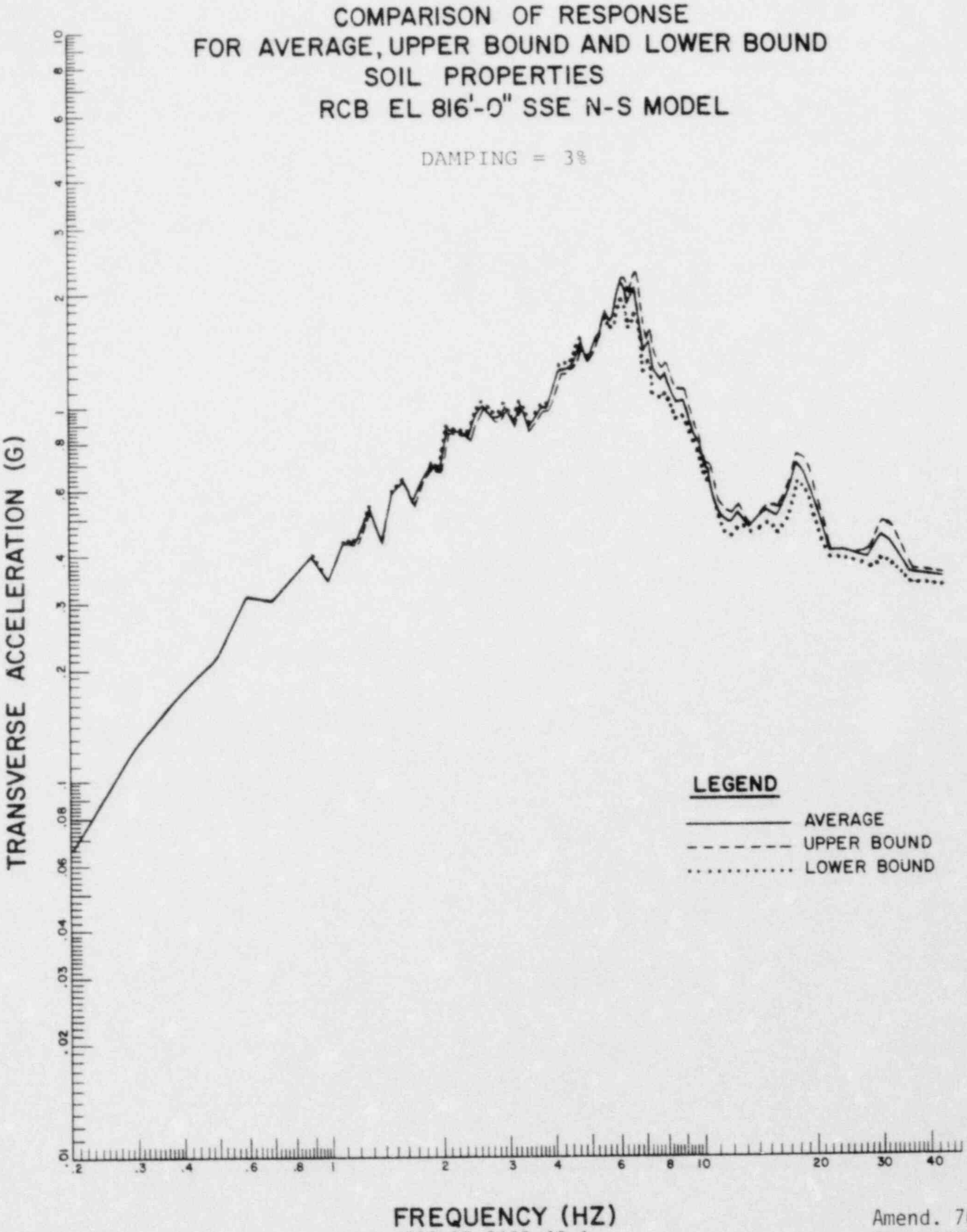

FIGURE Q220.42-1

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Question CS 220.43 (a)

The applicant used a finite element axisymmetric structural computer code to evaluate containment buckling from thermal stresses derived from other analysis. This is probably inappropriate because of the short wave lengths (local wrinkling effects) actually expected from thermal stresses in the shell. This local buckling results must be reconsidered in the applicant's evaluation of the containment shell in the PSAR.

Response

Thermal Buckling for CRBRP Containment Vessel. A linear elastic analysis was performed to compute the critical buckling temperature using the finite difference computer program BOSOR 4* (Ref. QCS 220.43(a)-1). This method is considered appropriate since the short wave length (local wrinkling) effects are accounted for. BOSOR4 results showed that the thermal buckling is of a local nature and occurs in the vicinity of Elevation 816".

The mathematical model has a cylindrical shell and a dome. The radius of the cylindrical shell is 93' and its' thickness is 1.5" from El. 816 to El. 849'.

The vessel was assumed clamped at Elevation 816 and a uniform, axisymmetrical temperature rise was assumed. The critical temperature (Tcrit) was computed using bifurcation buckling analysis which is treated as an eigenvalue problem by BOSOR 4. In the bifurcation buckling problem the eigenvalues (λ) are computed for several circumference wave numbers (n) and the stability equation is solved for the smallest eigenvalue. The value of (n) corresponding to the minimum λ is 57, i.e., there are 57 circumferential waves at the buckling temperature. This λ is used to calculate the value of Tcrit. The value obtained is Tcrit = 708°F, based on a reference temperature of 70°F. Therefore, the buckling temperature is 778°F. In addition to the computer calculations, a hand calculation was performed using the method of reference Q220.43(a)-2, which provides a procedure to determine the critical buckling temperature for cylindrical shells with various edge conditions.

The hand c[illegible]culation gave a value Tcrit = 1046°F for a hinged and cylindrical shell. [illegible] method of Reference QCS220.43(a)-2 gave a higher critical temper[illegible]e (Tcrit) for a 1-3/4 inch thick cylinder than for 1-1/2 inch. Since [illegible]SOR4 results are based upon a 1-1/2" thick vessel, while the revised thickness is 1-3/4", the computed Tcrit is on the conservative side. The temperature that causes yield at the bottom of the Containment Vessel is 270°F, which is substantially lower than the buckling temperature.

Based on this, the critical stress for thermal buckling can be considered to be yield stress.

References:

QCS220.43(a)-1 D. Bushnell - Stress, Stability and Vibration of Complex Branched Shells of Revolution: Analysis and User's Manual for BOSOR 4-NASA/Langley Research Center, Hampton, Virginia, March 1972.

QCS220.43(a)-2 D. J. Johns, "Local Circumferential Buckling of Thin Circular Cylindrical Shells" NASA TND1510 (1962).

*See Appendix to this response for details on BOSOR4.

Question CS 220.43(a) - Appendix

BOSOR4 COMPUTER PROGRAM

BOSOR4 is a computer program for stress, stability and vibration analysis of shells of revolution. The program was developed by D. Bushnell of Lockheed Missiles and Space Company.

The computer code is based upon the linear, elastic, thin shell theory. The structure should be axisymmetric. The program can handle various kinds of wall materials and loadings. Both mechanical and thermal loads are permitted in the analysis. In cases involving stress analysis of a shell for non-axisymmetric loading, the program finds the Fourier series for the loads, calculates the shell response in each harmonic to the load components with that harmonic, and superimposes the results for all harmonics.

The program has an option by which the stability analysis of a shell can be treated as a bifurcation buckling problem and mathematically it is treated as an eigenvalue problem. The program also handles shell vibration as an eigenvalue problem and finds mode shapes and frequencies.

BOSOR4 uses a finite-difference scheme as a numerical technique in the solution of shell problems.

Question CS 220.43 (b)

The applicant must consider the torospherical wrinkling caused by internal pressure in their containment evaluation in the PSAR.

Response

In some torospherical shells under internal pressure, compressive membrane stresses are developed; however, for the particular configuration of the CRBRP containment there are no such compressive stresses as shown in Table Q220.43(b)-1.

The configuration of the steel containment vessel dome is shown in Figure Q220.43(b)-1. It consists of a 1.25" thick ellipsoidal knuckle and 1-3/16" thick spherical cap. The specified steel is SA516 Grade 70. The design internal pressure is 10 psig.

Membrane forces N and N are shown in Figure Q220.43(b)-2 and

Table Q220.43(b)-1.

N_ϕ = Meridional membrane force in lbs/in.

N_θ = Hoop membrane force in lbs/in.

Sign Convention: Positive sign is tension, Negative is compression

The membrane forces clearly indicate that the dome is subjected to tensile stresses only under the internal pressure. Hence, stability of the dome is not a concern under the internal pressure and torospherical wrinkling will not occur.

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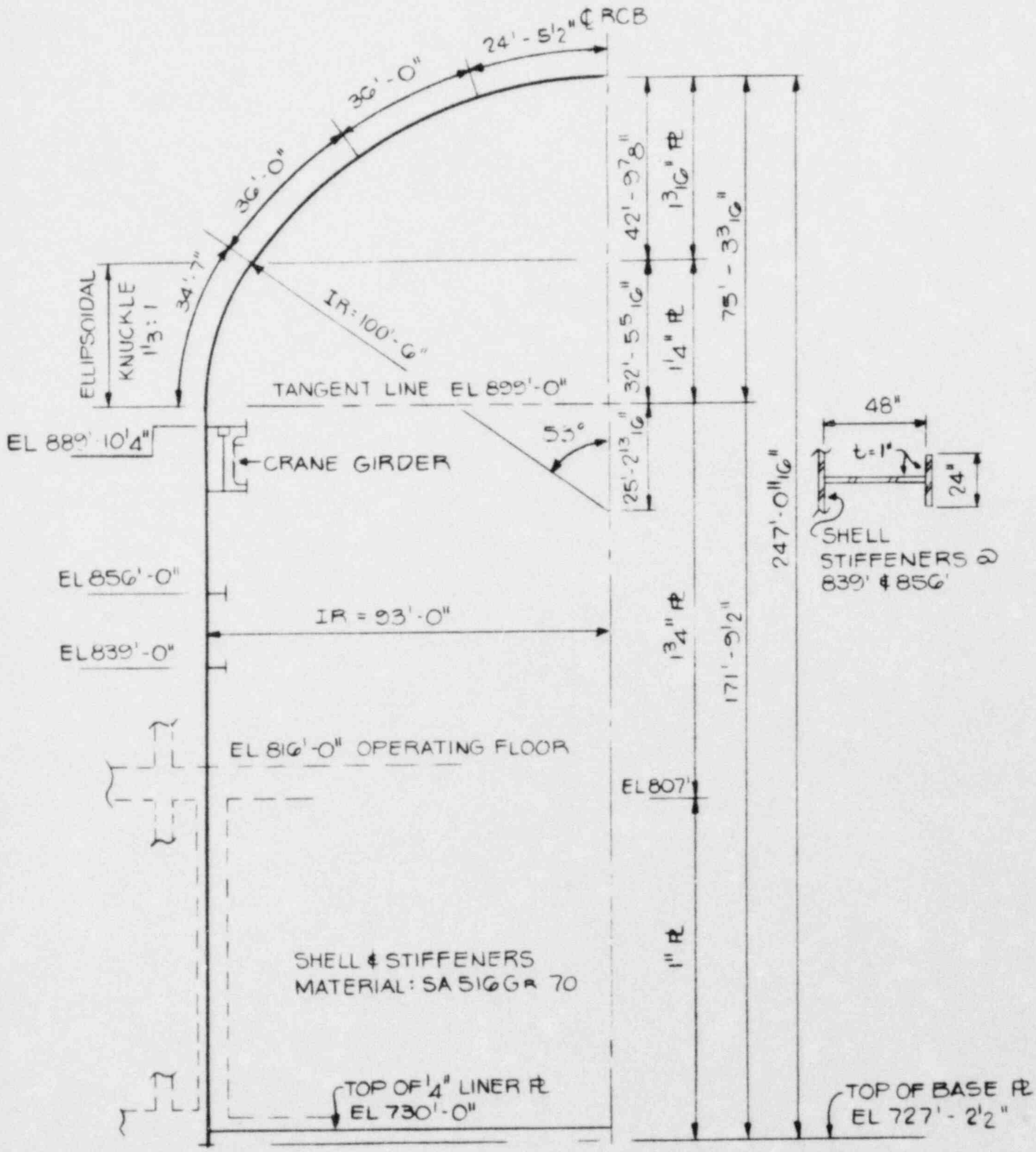
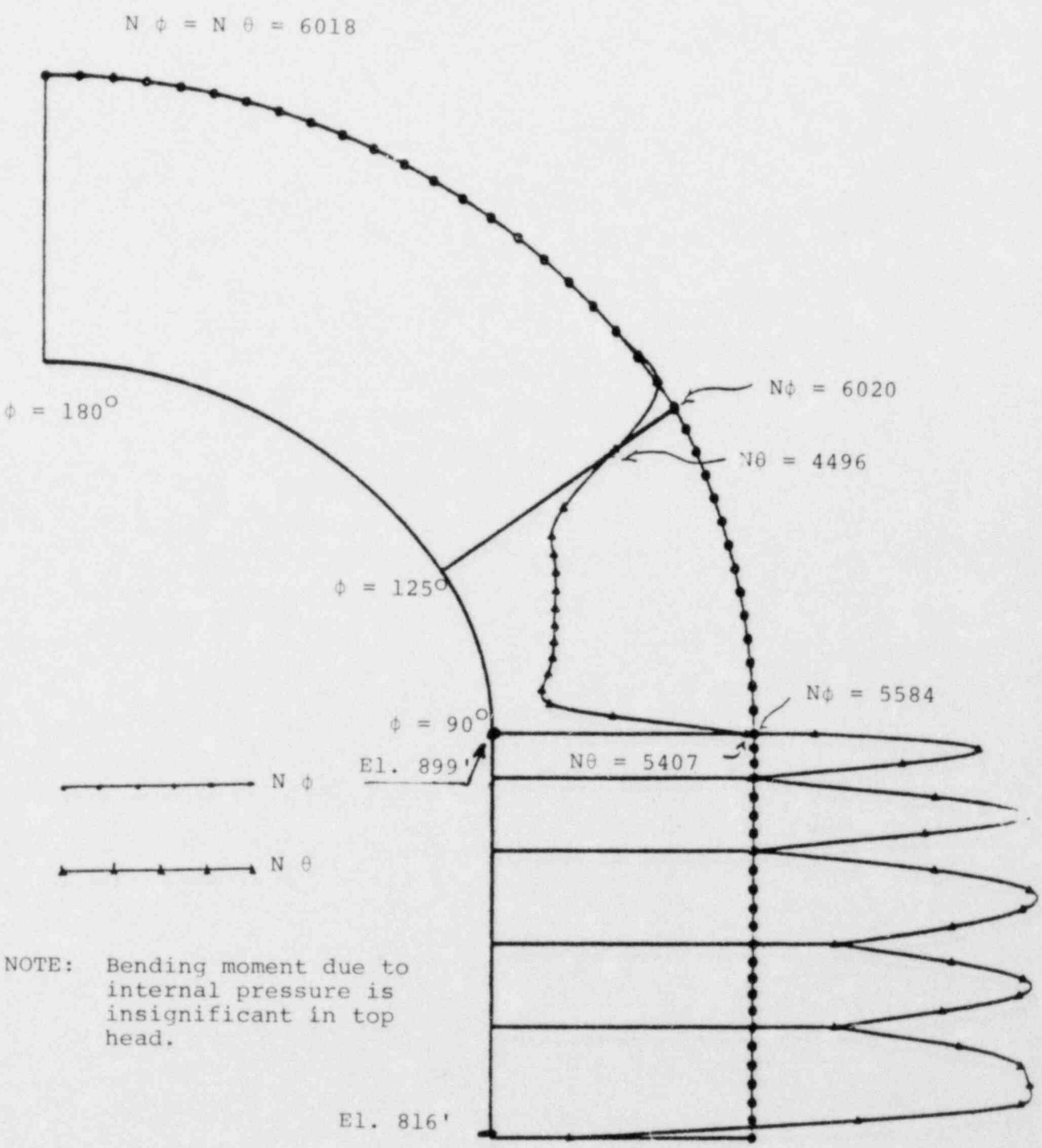


FIGURE Q220.43(b)-1 CRBRP CONTAINMENT VESSEL



NPHI, NTHETA ALONG MERIDION AT 0 DEG. (PSI)
CRBRP INTERNAL PRESSURE 10. PSI
MAXIMA Nφ = 6038. Nθ = 11659.

FIGURE Q220.43(b)-2 MEMBRANE FORCES Nφ AND Nθ DUE TO INTERNAL PRESSURE

TABLE Q220.43(b)-1

MEMBRANE FORCES IN DOME DUE TO 10 PSIG INTERNAL PRESSURE

Location OSE

COLUMN	1		2		3		4		5	
ø	DEAD LOAD + SCAF.CLIPS LBS/IN.		EXT. PRESS. .5 PSI LBS/IN.		INT. PRESS. 10 PSI LBS./IN.		SUM OF DEAD LOAD+EXT PRESS. COL 1&2 LBS/IN.		SUM OF DEAD LOAD+INT. PRESS COL 1&3 LBS/IN.	
	Nø	Nθ	Nø	Nθ	Nø	Nθ	Nø	Nθ	Nø	Nθ
90	-335	269	-279	-270	5584	5407	-614	-1	5249	5676
92.5	-324	469	-279.	-127	5586	2544	-603	342	5262	3013
95	-316	532	-280	-62	5592	1235	-596	470	5276	1767
97.5	-307	507	-280	-55	5604	1105	-587	452	5297	1612
100	-299	463	-281	-65	5621	1290	-580	398	5322	1753
102.5	-293	423	-282	-73	5642	1450	-575	350	5349	1873
105	-286	389	-283	-78	5667	1563	-569	311	5381	1952
110	-275	326	-287	-90	5732	1798	-562	236	5457	2124
115	-266	261	-291	-105	5815	2092	-557	156	5549	2353
120	-258	196	-296	-118	5918	2361	-554	78	5660	2557
125	-252	253	-301	-225	6020	4496	-553	28	5768	4749
130	-261	-22	-302	-304	6035	6078	-563	-326	5774	6056
135	-251	-52	-302	-302	6034	6033	-553	-354	5783	5981
140	-242	-85	-302	-302	6034	6034	-544	-387	5792	5949
150	-227	-139	-302	-302	6034	6034	-529	-441	5807	5895
160	-223	-184	-302	-302	6034	6034	-525	-486	5811	5850
170	-205	-195	-302	-302	6034	6035	-507	-497	5829	5840
175	-203	-201	-302	-302	6032	6037	-505	-503	5829	5836

NOTES: 1. Column 3 shows the distribution of membrane forces (Nø, Nθ) due to internal pressure in the dome.

2. ø is defined on Figure Q220.43(b)-2

Amend. 70
Aug. 1982

Question CS 220.43 (c)

The applicant has not evaluated buckling of large openings and buckling near local penetrations in the PSAR. We require this analysis be included.

Response

Buckling has been considered at large openings and near penetrations.

All penetrations are fully reinforced in accordance with ASME Code rules, therefore the buckling strength of the vessel will not be reduced by these penetrations. Of the two major penetrations above the operating floor, the equipment hatch was not only reinforced in accordance with the code but buckling is prevented by a stiffening frame provided around the opening as shown in Figure QCS 220.43(c)-1.

The other major penetration is the Equipment/Personnel Airlock and the stresses around the opening will be checked against the buckling allowable.

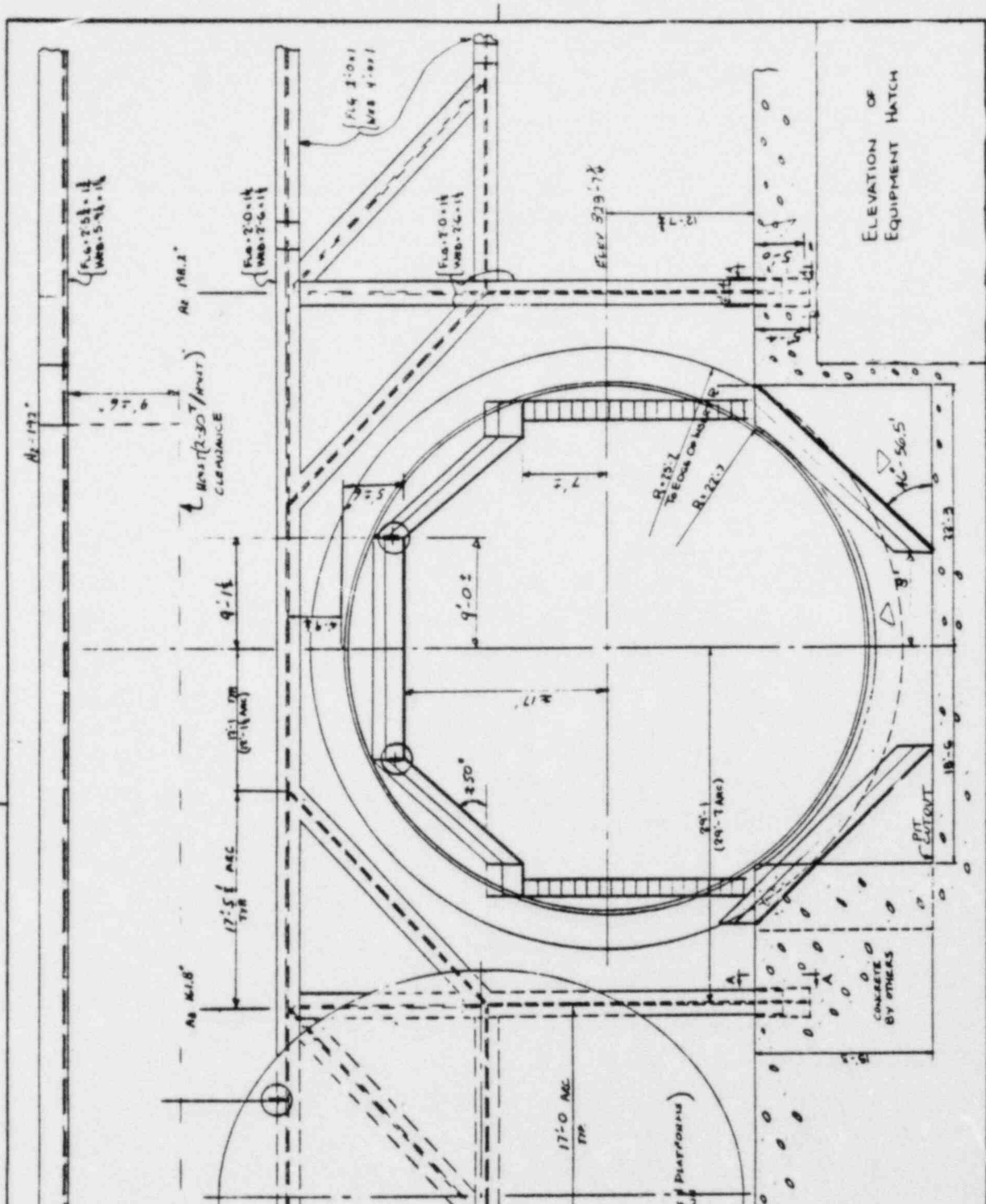


FIGURE Q220.43(c)-1

Amend. 70
Aug. 1982

Question CS 220.43 (d)

Provide a discussion on your analysis of the containment buckling in the region adjacent to polar crane support.

Response

The polar crane support consists of two ring stiffeners and 60 equally spaced vertical radial stiffeners or gussets as shown in Figures 1 Q220.43(d)-1 and Q220.43(d)-2.

The stresses on the shell including those from the Containment Vessel itself and those due to the polar crane reactions. The polar crane reactions were provided by the crane manufacturer and were applied as concentrated loads by means of Fourier Series on the shell using the Kalnin's program.

Buckling was checked in accordance with the buckling criteria given in PSAR Appendix 3.8-A.

The ring stiffeners were checked in accordance with new Section 7 of Appendix 3.8-A.

The vertical stiffeners (gussets) were checked as column sections per the AISC code.

The critical buckling stresses of the shell plates between the ring stiffeners and vertical gusset plates were calculated using the equations presented in Sections 3.2.a, 3.2.b, 3.2.c and 3.2.d of the PSAR Appendix 3.8-A, for axial compression, circumferential compression, torsion and bending respectively. The critical buckling stress for transverse shear was made equal to 1.25 the critical shear stress for torsion per Section 6.0 of the PSAR Appendix 3.8A.

Safety factors of 1.33 for the OBE and 1.11 for the SSE were used in accordance with Table 3.8-A-2 of the PSAR Appendix 3.8A.

Based on the calculated shell stresses, critical buckling stresses and safety factors, the stability was evaluated through the interaction equation in revised Section 6.0 of the PSAR Appendix 3.8A and the results were shown to meet the acceptance criterion.

Amend. 70
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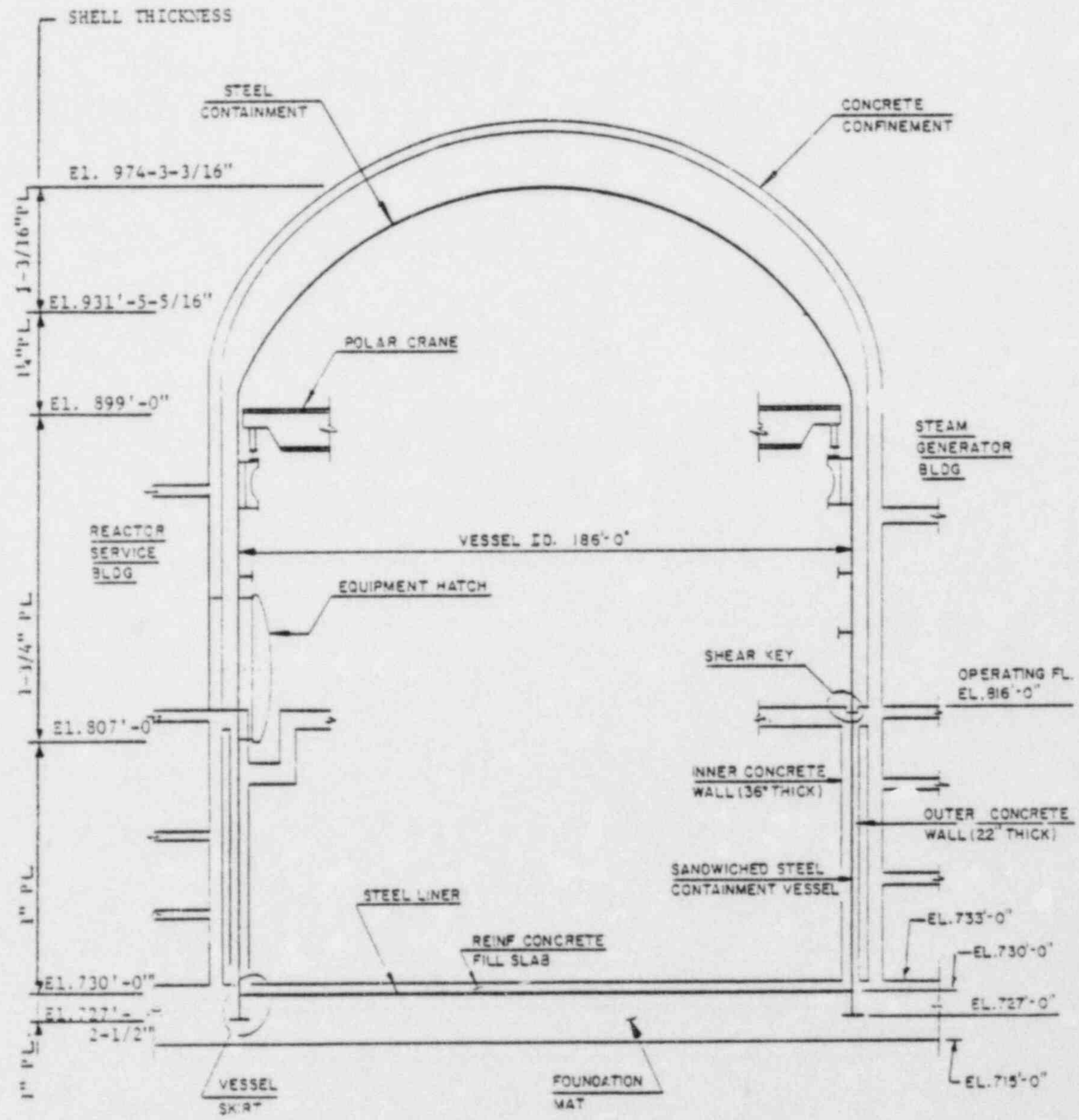

FIGURE Q220.43(d)-1 CONTAINMENT BUILDING CROSS SECTION

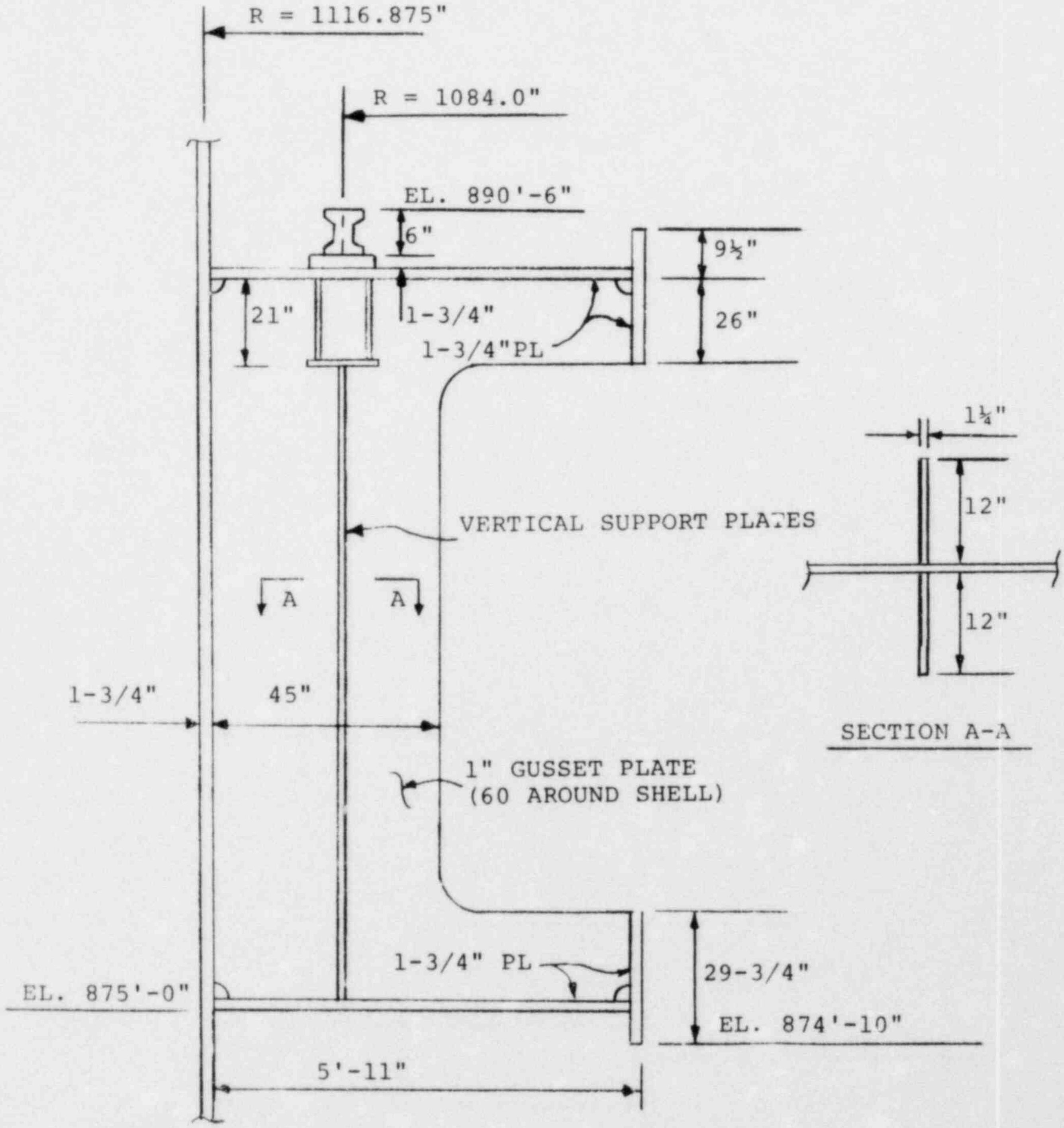

FIGURE Q220.43(d)-2 SECTION THRU CRANE GIRDER SUPPORT

Question CS220.44

We require additional detail for the reactor vessel support ring before evaluating that structure. In particular, the means for transferring load from the ring to primary concrete structure needs to be provided. The applicant should also provide their analysis of this load transfer path and the predicted safety margin.

Response:

Additional details for the Reactor Vessel Support Ring are provided in response to Question CS220.34(a).

The Reactor Vessel Support Ledge (RVSL) is a steel structure supported by, and embedded in, the Reactor Cavity as shown in PSAR Figure 3.8-9.

A system of horizontal ring plates, radial brackets and stiffeners, and vertical cylindrical panels welded together, comprise the Reactor Vessel Ledge.

The two upper ring plates, at Elevation 800'-7-1/4" and at Elevation 795'-1-1/4", receive the loads directly from the Reactor Vessel Support System.

The upper support ring plate receives the downward load by bearing. The lower support ring plate takes the upward loads through hold down bolts. (The holddown bolts transfer the upward load to the lower support ring plate).

The base plate (El. 780'-7-1/4") is embedded in concrete and is also anchored in the concrete by anchor bolts.

The exterior cylinder (R=20'), together with the radial brackets and stiffeners, transfer the vertical loads from the support plates to the base plate. The two inner cylinders (at R=12'-0-1/2" and at R=13'-11-1/2") transfer loads and contribute to stiffen the intersecting plates by forming box sections.

The radial brackets continue outwards (beyond the exterior cylinder at R=20') to constitute the radial stiffeners. These radial stiffeners are embedded in the cavity concrete and act as shear keys for horizontal loads.

Thus the vertical loads, upward and downward, are first taken by the upper and lower plates respectively. These are then transferred to the base plate by (1) the radial plates and (2) the exterior vertical cylinder. The downward vertical loads are transferred to the concrete by the base plate by bearing. The upward vertical loads are transferred by tension in the base plate anchor bolts and shear in the concrete above the base plate.

The horizontal loads (seismic) are transferred through the upper horizontal plates, and the radial brackets to the stiffener plates embedded in the concrete acting as shear keys. The load is transferred to the concrete by bearing against the stiffener plates.

The 1/2 inch radial gap between the exterior cylinder and the concrete, allows for free thermal expansion of the ledge under DBA conditions.

The most severe loading on the ledge is the SMBDB load. (see Response to Question CS220.34(b)). Project assessments show that the support rings can accommodate the full SMBDB load.

Question CS 220.45

From your presentation on seismic analysis it appears that in your mathematical model, you used the 2-D finite element analysis to derive the spring constants for soil and used damping values on the basis of half space theory. The staff has reservations in such an analysis approach and your justification for such an approach is requested.

Response

The use of half-space theory is an accepted method to determine spring and damping constants for soil-structure interaction. If, instead of using the theoretical equations, the half-space is represented by finite elements, the calculated spring constants will be the same (or close) as the theoretical values. Since the material under the CRBRP foundation mat is not strictly a half-space because of the inclined layers of siltstone and limestone, a static finite element approach was used. The purpose was to define an equivalent half-space. The rock under the foundation is not considered sufficiently rigid to jusfify a fixed base analysis.

Although a two-dimensional finite element analysis was used, a correction factor for three-dimensional effects was introduced as indicated in Section 3.7.1.6 of the PSAR (Page 3.7-3b).

The spring constants calculated in this manner were compared with the values calculated using the half-space equations (using siltstone as uniform material under the foundation) and the agreement between the two sets of values was satisfactory.

The damping values for the soil-structure interaction were calculated with the theoretical half-space equations. This was further justified by the satisfactory agreement between the finite element calculated springs and half-space equations.

In addition, the analyses were performed for upper bound, lower bound and average rock properties and the results for the three cases were enveloped. This would account for any uncertainties in properties and methods of calculations.

Question CS 281.6

In the CRBR primary and intermediate sodium piping system, Fe, Cr, and N:I are dissolved from the high temperature regions and deposited in the lower temperature regions because of super-saturation. Included in this process of mass transfer is the formation and decomposition of various transition metal and sodium double oxides. Deposition of these mass transfer and corrosion products may cause flow restrictions and loss of heat transport efficiency of heat exchangers. Describe the criteria and bases in your analyses of mass transfer and deposition of corrosion products in the CRBR primary and IHX sodium systems to assure necessary system flow and heat transfer. Include the instrumentation and detection system which will alarm when these limits are exceeded.

Response:

Primary Sodium Piping System

The IHX is designed with an effective tube length of 24.21 feet which is 5.12 ft or 33% greater than required for nominal operation. Included in the 33% excess allowance is a 9% factor for heat transfer degradation due to the deposition of mass transfer products on the primary side of the unit.

Corrosion products go into solution in the core region of the reactor either by direct dissolution or by the formation of soluble oxide complexes. As the coolant flows through the cooler regions of the system it becomes super-saturated with respect to these corrosion products and they precipitate out. Precipitation is expected to occur in the IHX. The deposits will result in some degradation of the overall heat transfer coefficient. This potential problem was recognized in the early 70's and work was performed to determine the magnitude of this effect A summary of this work is given below.

Two tube-in-shell heat exchangers were available from an ongoing corrosion program. The first had operated in an all stainless steel system with a hot leg temperature of 1325°F for 0.84 years. The second operated in a Type 304ss/Incoloy 800 system for 1.5 years with a hot leg temperature of 1100° and a high, (28 ppm by amalgamation) oxygen level. Heat transfer measurements were made on these heat exchangers and compared with similar measurements on new heat exchangers of identical design. The percentage change in heat transfer coefficient was determined for each set of readings.

The exposure conditions experienced by the heat exchangers were excessive in that the first one operated with a high maximum hot leg temperature (1325°F) and the second with a high oxygen level. Equivalent operating times at reactor operating conditions (1100°F mean T_{max}) were calculated at 5.3 years and 11.4 years. It was judged that the deposit thicknesses reached equilibrium. Additional increases in thickness are prevented by flow induced shearing of the friable deposits.

The average deposit heat transfer resistance was calculated as 8.4×10^{-5} h-ft^2 °F/BTU. The overall heat transfer coefficient of the IHX design was 1190 BTU/h-ft^2 °F. Adding the deposit resistance gives a calculated degradation of 9%. This 9% value was used for the FFTF IHX design and in the CRBRP design.

The effect of corrosion products on pressure drop and flow blockage is not addressed in section 5.3 of the PSAR. Flow blockage is addressed in section 15.4.1.3. Flow blockage in the Core Assembly in 15.4.1.3.1. 'Prevention and Detection' C, (Corrosion Products).

Deposition induced pressure drop in the IHX is not considered to be a factor because of the large flow cross sections on the shell side.

The effects of corrosion product deposition in the PHTS will result in a very gradual change, if any, in system performance. The PHTS performance is continuously monitored and critical performance parameters are calculated by the plant computer from temperature, flow rate and pressure drop sensors in the PHTS system.

Monitoring sensors in each loop include resistance temperature detectors at the inlet and outlet of the IHX, pressure sensors in the hot and cold legs, and a flow meter.

The performance evaluation for the PHTS is identified in the system procedures and includes: 1) IHX Thermal Performance - calculation of overall U value, 2) Total Loop Pressure Drop, 3) IHX Primary Side Pressure Drop, and 4) Reactor Vessel Pressure Drop.

Intermediate Sodium Piping System

The Intermediate Heat Transport System is designed to operate with the oxygen content in th sodium controlled to 2.0 PPM maximum to enhance the leak detection capability in the system. A leak in a water/steam tube of a plant evaporator or superheater results in an oxygen concentration increase in the intermediate sodium; thus low oxygen background is maintained for early detection should such a leak occur.

The 2.0 PPM oxygen level is an order of magnitude lower than that required for controlling corrosion of the austenitic stainless steel in sodium. Therefore, the mass transport of corrosion products and the subsequent potential degradation of heat transfer capacity is not a concern in this system with this sodium coolant chemistry. Furthermore, in the steam generators, which are in the cooler sodium region of the system and are, therefore, most subject to mass transfer deposition, the limiting heat transfer conditions exist on the steam/water side of the tubes which would not be affected by sodium chemistry.

The intermediate sodium oxygen level is monitored by an on-line plugging temperature indicator and periodic sodium samples are also obtained. See PSAR Section 9.8.