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SUBJECT: Forwards Amend 1 to updated FSAR. RGU 8/4/83 RB

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JUL 20 1983

E. E. UTLEY
Executive Vice President
Power Supply and Engineering & Construction

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
SUBMITTAL OF UPDATED FSAR AMENDMENT 1

Dear Mr. Denton:

In accordance with the requirements of 10 CFR 50.71(e), Carolina Power & Light Company (CP&L) hereby submits one (1) original and twelve (12) copies of Amendment 1 to the updated Final Safety Analysis Report (FSAR) for the H. B. Robinson Steam Electric Plant (HBR) Unit No. 2.

Amendment 1 to the updated FSAR incorporated changes made under the provisions of 10 CFR 50.59 but not previously submitted to the Commission. These changes have been appropriately located within the updated FSAR.

The updated FSAR was submitted on July 20, 1982, with data, drawings, and figures current through January 22, 1982. Amendment 1 to the updated FSAR accurately presents changes made since the updated FSAR submittal necessary to reflect information and analysis submitted to the NRC or prepared pursuant to NRC requirements. Amendment 1 to the updated FSAR is current through January 22, 1983.

Yours very truly,

E. E. Utley

SHC/cfr (7395PUB)
Attachment
cc: Mr. J. P. O'Reilly (NRC-II)
Mr. G. Requa (NRC)
Mr. S. Weise (NRC-HBR)

E. E. Utley, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and that the sources of his information are officers, employees and agents of Carolina Power & Light Company.

Franklin Murray
Notary (Seal)

My commission expires: October 4, 1986

FRANKLIN MURRAY NOTARY PUBLIC WAKE COUNTY, N. C.

A053 1/12

Superseded PSS PGR REV 1 + updated FSAR 50-261 7/20/83 8308010295

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APPENDIX 1.1A
REDUCED TEMPERATURE OPERATION

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APPENDIX 1.1A REDUCED TEMPERATURE OPERATION

1.1A.1 INTRODUCTION

As a temporary measure to improve the operating conditions within the steam generators, H. B. Robinson (HBR) 2 is being operated at reduced levels of temperature and power. The major changes are that the reactor coolant inlet, outlet and average temperatures are all reduced, the steam generator secondary side temperature and pressure are reduced, reactor power is reduced, and the turbine-generator is operated with its governor valves at or near the fully opened position. The changes are described below, and additional information is given in References 1.1A.1-1 through 1.1A.1-3.

1.1A.2 OPERATING CONDITIONS

Table 1.1A.2-1 compares normal operating conditions to the reduced temperature operating conditions. Operation in this mode requires revisions to the protective system setpoints, as discussed in the Technical Specifications (Reference 1.1A.1-3). The specific reactor trip setpoints which require revision are the high flux power range, the overtemperature ΔT and the overpower ΔT.

Engineered Safeguards initiation circuits which require setpoint changes are those which provide steam line break protection by initiating safety injection (SI) and steam line isolation on 2/3 high steam flow with low T_{avg} or low steam pressure (Table 7.3.1-1).

The reactor trip setpoint changes are in the conservative direction, in that reactor power and ΔT are limited to lower valves than for normal operation. The steam line break protection setpoints on low ΔT and low steam pressure are reduced to lower valves consistent with the reduced operating values.

TABLE 1.1A.2-1

COMPARISON OF NORMAL TO REDUCED POWER OPERATING CONDITIONS

	NORMAL OPERATION	REDUCED TEMPERATURE OPERATION
Reactor Coolant Temperature at Zero Reactor Power, °F	547	530
Maximum Rated Reactor Power, Percent of 2300 Mwt	100	≈81
Hot Leg Temperature at Maximum Rated Reactor Power, °F	605	560
Cold Leg Temperature at Maximum Rated Reactor Power, °F	546	513
Average Coolant Temperature at Maximum Rated Reactor Power, °F	575	537
Steam Pressure at Maximum Rated Reactor Power, psig	800	580
F_Q	2.2	2.2

1.1A.3 SAFETY ANALYSIS

1.1A.3.1 Loss-of-Coolant Accident

An analysis was performed to determine the effects of operating HBR Unit 2 at reduced temperature on calculated loss-of-coolant accident (LOCA) results. This analysis included previous calculations for HBR and calculations for a plant similar to HBR that was analyzed for a reduced temperature program. To provide operational flexibility at the expected maximum power level (approximately 81 percent of rated power) the analysis was performed assuming 85 percent of rated power. The conclusions of this analysis were that reduced temperature operation at 85 percent rated power with the currently allowed F_Q value of 2.2 will lead to enhanced LOCA Emergency Core Cooling System (ECCS) margins.

In the qualitative evaluation of the effects of reduced temperature and power operation for HBR 2, a bounding peak cladding temperature (PCT) increase of 200^{o}F was given due to the reduced temperature effect. This number was given based on unreported results obtained for a similar Westinghouse two-loop pressurized water reactor (PWR). The reason for the PCT increase is the reduced primary system temperature or enthalpy which reduces the LOCA energy release to the containment. This in turn reduces containment pressure and reflood rates, and thereby increases calculated PCT. The two-loop PWR results, which were performed at an assumed 100 percent power, are believed to be suitable to bound the reduced primary temperature effects on HBR because of the similarity of calculated results for both reactors, particularly with regard to containment pressure and reflood rate. The calculated results for containment pressure and reflood rate for the two reactors, shown in Figures 1.1A.3-1 through 1.1A-3-4, demonstrate this similarity.

The 200^{o}F PCT increase for the two-loop PWR is the difference between the low temperature results and the current licensing analysis. About 45^{o}F of this increase is associated with incorporation of the ENC WREM-IIA model. A 155^{o}F PCT difference between the two cases using the same analytical models was calculated to occur on a ruptured node with enhanced metal-water reaction.

The PCT increase at the node corresponding to the non-ruptured PCT location is about 80^{o}F. A similar PCT increase (~80^{o}F) would be expected for HBR, which due to reduced power, will not become rupture node limited. The more conservative 200^{o}F difference was chosen to assure that any system dissimilarities are bounded.

For HBR 2 the 300^{o}F PCT decrease due to power reduction stems from the assumed 15 percent reduction in total core power while maintaining the 2.2 peaking factor, which assures a 15 percent reduction in allowed linear heat generation rate. Exxon Nuclear Corporation (ENC) has performed several LOCA-ECCS calculations for HBR 2 at various assumed values of power peaking and steam generator tube plugging. The variation of PCT with local power (20-25^{o}F per percent local power) was obtained directly from these results. The 300^{o}F decrease is a minimum number resulting from the 15 percent power reduction and the minimum value of the range. On an average basis, the benefits of reduced power are four times the best estimate of reduced temperature effects and a significant reduction in PCT (1922^{o}F based on latest ENC analysis) is expected for HBR 2 operating at the reduced power and temperature.

1.1A.3.2 Anticipated Transients

Anticipated operational transients at the proposed operating conditions and RPS setpoints were also reviewed. As part of this review, thermal hydraulic calculations have been performed which show a substantial increase in the initial minimum departure from nucleate boiling ratio (MDNBR). This increase in initial MDNBR assures increased thermal margins for anticipated operational transients, since the changes in MDNBR during these transients will not be substantially affected from those previously evaluated. This review specifically addressed those FSAR Chapter 15 transients previously demonstrated to be most limiting with respect to thermal margin. These transients include the reactor coolant pump coastdown and locked rotor loss-of-flow transients, rod withdrawal transients, and the large steam line break accident. The other transients indentified in the FSAR will continue to remain nonlimiting for the reduced temperature and power operating conditions.

1.1A.3.2.1 Steady State Thermal Margin

The steady state thermal margin calculations employed the conditions given in Table 1.1A.3-1 and allowed for 12 percent overpower, 4.5 percent bypass flow, and measurement uncertainties as noted. The calculated MDNBR is 2.83, indicating very substantial thermal margin at the reduced power, temperature, and flow conditions. The reference calculation (Reference 1.1A.3-1) employed rated full load (1.12 x 2300 Mwt) operating conditions and yielded an MDNBR of 1.87. The increased margin at the reduced power conditions results from a 15 percent reduction in clad surface heat flux and a 25 percent increase in inlet subcooling relative to the rated full load condition. The margin to DNB at the reduced power condition is about twice that calculated for rated full load conditions.

1.1A.3.2.2 Effect of Reduced Moderator Temperature on Hot Channel Peaking During the Rod Withdrawal Transient

In the uncontrolled rod withdrawal at power transient, reduced primary coolant temperature should have an insignificant impact on power peaking magnitude when it is compared to the normal temperature case. Because the change in coolant temperature affects the entire core and power density calculations are relative, the power peaking magnitude should remain essentially the same at different primary coolant temperatures.

Nevertheless, calculations have been performed to quantify the effect of reduced moderator temperature on hot channel peaking during the rod withdrawal transient. These calculations indicate that the reduced moderator temperature under the new temperature program results in about a 3 percent increase in peaking augmentation caused by an inadvertent control rod withdrawal. This effect is judged to be insignificant in itself and is certainly more than offset by the 15 percent reduction in the high nuclear flux and overtemperature ΔT reactor trip setpoints which will be in effect under the reduced operating temperature conditions.

The moderator temperature coefficient for the remainder of Cycle 8 is calculated to be between -10 pcm/^{o}F and -32 pcm/^{o}F, well below the Technical Specification limit of +2 pcm/^{o}F.

1.1A.3.2.3 Loss of Normal Feedwater Transient

Of all transients involving the loss of secondary side heat removal capability, the Loss of Normal Feedwater transient is the most significant. The key assumptions in the FSAR analysis of this transient are: (1) Immediate Reactor Trip, (2) Auxiliary Feedwater availability, (3) Reactor Power at 102 percent. The new reduced temperature program will not affect the first two assumptions, but will result in a lower Reactor Power. Therefore, the consequences of a Loss of Normal Feedwater transient under the reduced temperature program will be more conservative than analyzed in the FSAR.

TABLE 1.1A.3-1

CONDITIONS USED IN THE STEADY STATE THERMAL MARGIN CALCULATIONS

CORE CONDITIONS	NOMINAL	MARGIN ANALYSIS*
Power Level, Mwt	1955	2233*
Primary System Pressure, psia	2250	2220
Coolant Flow Rate, lbm/hr	98.0	93.58**
Coolant Inlet Temperature, ^{o}F	510	514***
T_{avg} SCHEDULE		
T_{avg} at No Load, ^{o}F		530
Linear Gain, ^{o}F/Percent Power		0.0935
POWER PEAKING FACTORS		
$F^{N}_{\Delta H}$		1.55
Axial Peaking Factor		1.55
Engineering Heat Flux Factor		1.03
Total Peaking Factor		2.47
Fraction of Power Deposited in the Fuel		.974
RESULT		
MDNBR, at Overpower, for Reduced Power and Temperature		2.83
MDNBR, at Overpower, for Rated Conditions (Reference 1.1A.3-1)		1.87

* Steady state overpower thermal margin analysis: 12 percent overpower allowance, and 2 percent measurement uncertainty: 1955 x 1.02 x 1.12 = 2233

** Analysis value assumes 4.5 percent bypass flow

*** Analysis allows for 4^{o}F deadband and measurement uncertainty

1.1A.4 OTHER CONSIDERATIONS

In addition to the Technical Specification changes and other items addressed above, the following items have been addressed by CP&L and its contractors in developing the proposed reduced temperature program:

a) Moisture carryover for the steam generator is not expected to exceed the .25 percent limit. Modifications made to the HBR 2 steam generators in 1980 have reduced the moisture carryover to .04 percent at full power, which is significantly below the required limits. This reduction in carryover is expected to more than compensate for any increase in moisture carryover as a result of the reduced temperature program. In fact, some preliminary calculations show that the moisture carryover may actually be reduced under the proposed operating conditions.

b) The new reduced temperature program will result in a higher ΔP (ΔP current = 1400 psia, ΔP proposed = 1620 psia) across the steam generator tube sheet, due to the reduced secondary system pressure. The HBR steam generator problems are due predominantly to temperature related effects and not mechanical stresses, therefore, a reduction in primary pressure is not necessary as a result of the small increase in mechanical stress.

c) No changes to the secondary water chemistry have been recommended as a result of the proposed reduced temperature program. (HBR 2 has phosphate secondary chemistry.)

d) It is not anticipated that there will be any significant impact on turbine disc cracking from the proposed reduced temperature operating conditions. The new operating conditions will require the turbine governors to be fully open. This will decrease the current pressure drop across the partially closed governor valves which will partially compensate for the lower steam line pressure. In addition, the turbine inspection performed last refueling outage (Fall, 1980) showed no indications in the low pressure turbine disks.

e) Relative to the reactor vessel integrity issue, the proposed reduced temperature program will result in slightly lower temperatures at the reactor vessel beltline. This reduced temperature could reduce self-annealing of the reactor vessel, however, this effect is expected to be minimal and no noticeable change in the rate of RT_{NDT} increase is anticipated during the short period of time (couple of years) during which the reduced temperature program is expected to be in effect. In addition, reactor power will be at a reduced level (approximately 81 percent) which should more than compensate for any reduction in self-annealing.

f) Another compensating factor is that overcooling transients are expected to be less severe at the reduced temperature conditions due to the expected smaller rate of temperature decrease. The effect of reduced temperature operation along with long-term corrective measures will be addressed within the on-going Reactor Vessel Integrity Program.

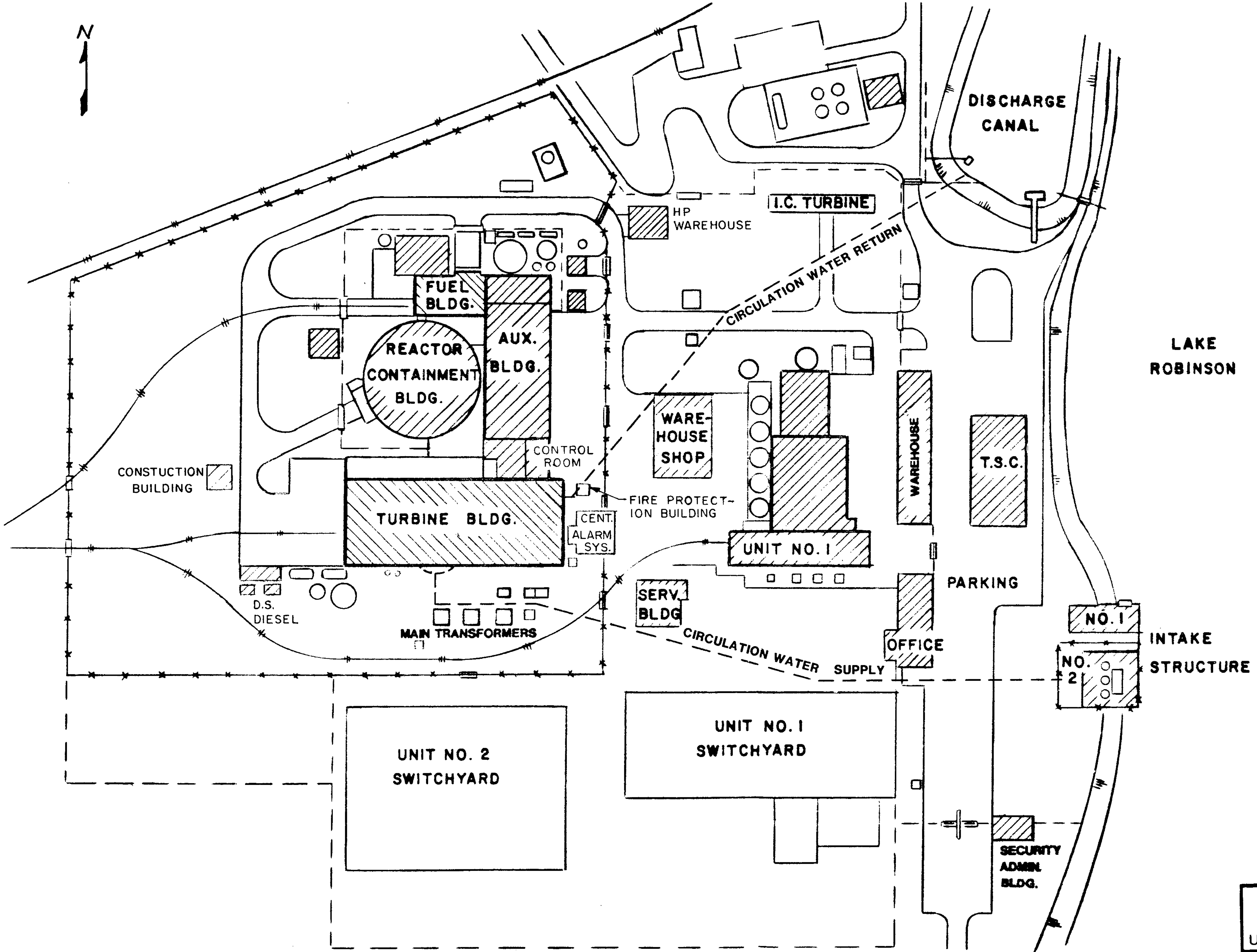
REFERENCES: APPENDIX 1.1A

1.1A.1-1 Letter, NO-81-1853, Nov. 11, 1981, from CP&L (Utley) to NRC (S. Varga), subj: H. B. Robinson Steam Electric Plant Unit 2, Docket No. 50-261, Lic. No. DPR-23, Request for License Amendment - Reduced Primary Coolant Temperature Operation, w/enclosures.

1.1A.1-2 Letter, NO-81-1983, Dec. 2, 1981, from CP&L (Zimmerman) to NRC (S. Varga), subj: H. B. Robinson Steam Electric Plant, Unit 2, Docket No. 50-261, License No. DPR-23, Correction to Request for License Amendment Reduced Primary Coolant Temperature Operation.

1.1A.1-3 Letter, November 13, 1981, from NRC (S. Varga) to CP&L (Jones), with Amendment No. 61 to Facility Operating Lic. No. DPR-23 for H. B. Robinson Steam Electric Plant, Unit No. 2 (revising Tech Specs).

1.1A.3-1 Report XN-75-25, "H. B. Robinson Fuel Design Report Volume 1 - Mechanical and Thermal Hydraulic Design for Cycle 4," June 1975.



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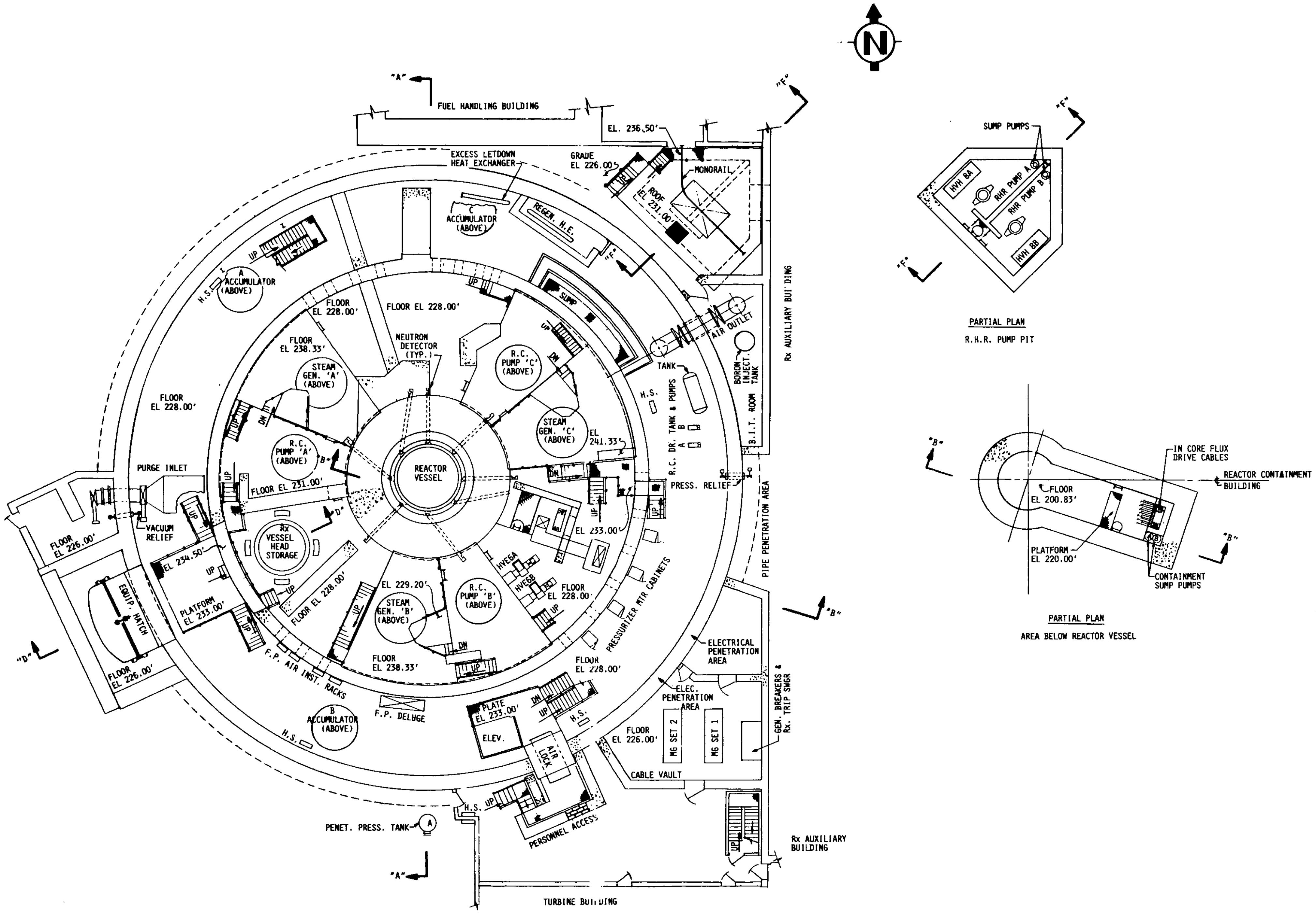
PLOT PLAN

FIGURE 1.2.2 - 1



REACTOR CONTAINMENT BUILDING PLAN - GROUND LEVEL

FLOOR ELEVATION 228'

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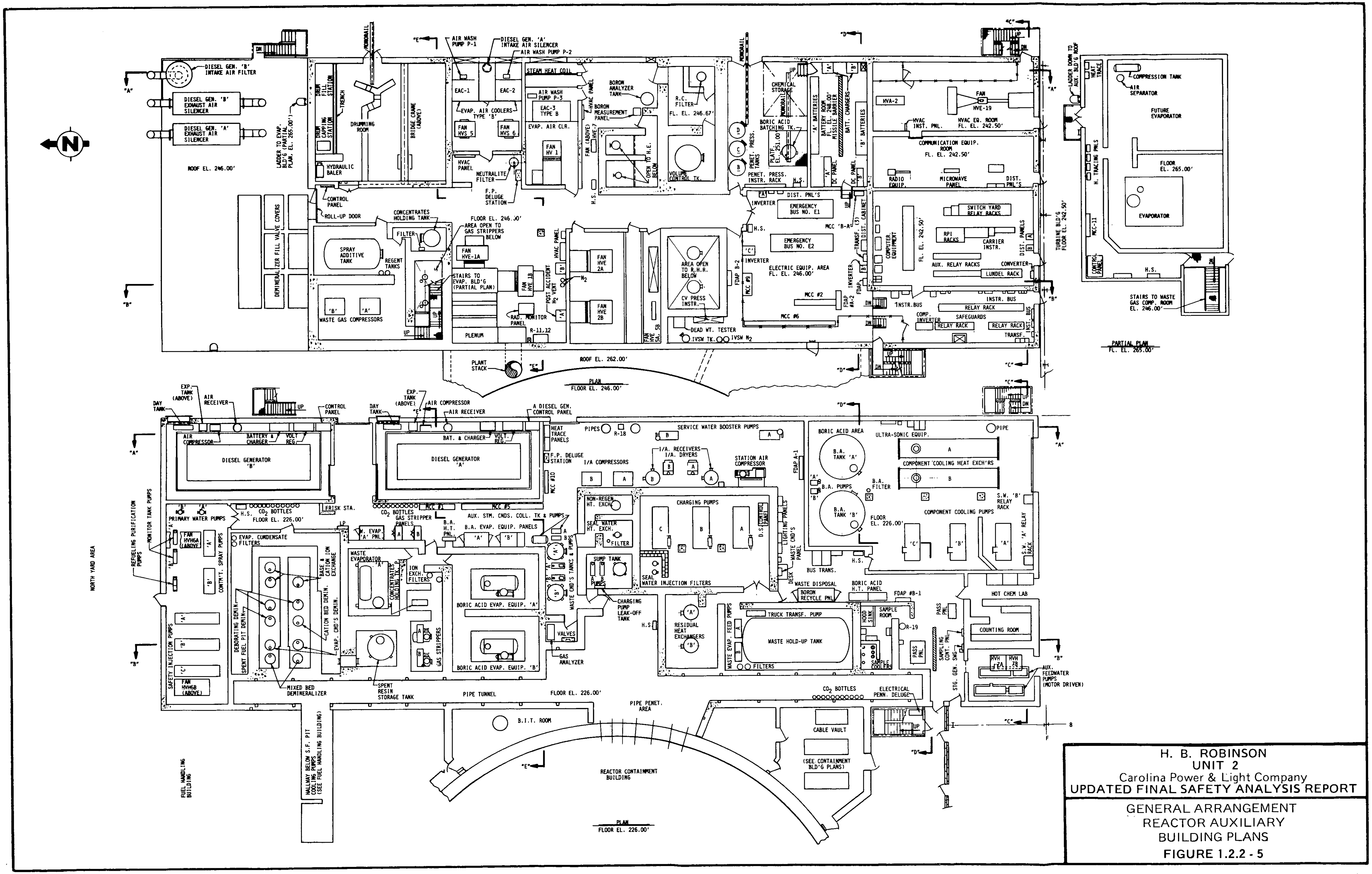
GENERAL ARRANGEMENT
REACTOR BUILDING PLANS
SHEET 1

FIGURE 1.2.2-2



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GENERAL ARRANGEMENT
REACTOR AUXILIARY
BUILDING PLANS

FIGURE 1.2.2 - 5

1.5.3 SYSTEMS FOR REACTOR CONTROL DURING XENON INSTABILITIES

In the transition to large Zircaloy-clad-fuel cores, the potential of power spatial redistribution caused by instabilities in local xenon concentration was created.

Extensive analytical work has been performed on reactor core stability (References 1.5.2-1, 1.5.3-2, and 1.5.3-3). These references indicate that a core of this size may be unstable against axial power redistribution, but is nominally stable against transverse (denoted X-Y) power oscillations. The plant is therefore provided with instrumentation which will allow the operator to detect the axial power oscillations, and procedures exist for suppressing these oscillations (Section 7.7.1).

Control information for suppression of power oscillation is obtained from four long ion chambers, each divided into an upper and lower section, mounted vertically outside the core. Both calculation and experimental measurements at SENA, San Onofre and Haddam Neck have shown that this out-of-core instrumentation represents in core power distribution which is adequate for power distribution control (Reference 1.5.3-3).

The control strategy is based on the difference in output between the top and bottom sections of the long ion chambers. If the operator allows axial power imbalance to exceed operating limits, various levels of protection are invoked automatically. These include generation of alarms, turbine power cutback and blocking of control rod withdrawal. The Axial Power Distribution Monitoring System (APDMS) used to monitor power oscillations is discussed in Section 7.7.1.

Regulatory Guide 1.39 — HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS (REVISION 0)

The applicable requirements of N45.2.3-1973 are followed at HBR 2 within the context of the established QA Program with the following specific exception -- the zone designations of Section 2.1 of N45.2.3 and the requirements associated with each zone are considered impractical for implementation, as stated, at HBR 2 during the operations phase. Instead, procedures or instruction for housekeeping activities, which include the applicable requirements outlined in Section 2.1 of N45.2.3 and which take into account radiation control considerations, security considerations, and cleanliness requirements are developed on a case basis for work to be performed.

CHAPTER 2

SITE CHARACTERISTICS

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SITE CHARACTERISTICS

LIST OF FIGURES

2.4.7 GROUND WATER

The drainage area of the Black Creek above the dam site is underlain and bounded by the Middendorf Formation, a sequence of unconsolidated and semiconsolidated cross-bedded, micaceous, feldspathic quartz sand and gravel beds.

These beds are intercalated with clayey sands and impure clays and lenses of white kaolin. These kaolin lenses can extend laterally for quite some distance and have a maximum thickness of approximately 35 ft. In most cases they are responsible for the existence of perched ground water in overlying sands. The sand and clay beds of the Middendorf are lenticular and grade laterally into one another or pinch out within comparatively short distances. This complex interfingering precludes the tracing of any particular bed for any great distance.

The Middendorf is a permeable Formation, and in several areas of the Coastal Plain yields up to 2000 gpm from individual wells. Ground water occurs under both water table and artesian conditions. In the former, the water surface is unconfined (under atmospheric pressure) and is free to move in a vertical direction. Under artesian conditions, the water in the aquifer is confined under a relatively impermeable bed and hydrostatic pressure causes the water to rise above the bottom of the confining bed when the aquifer is penetrated or exposed to the surface. Water in shallow aquifers is generally unconfined and in deeper aquifers it is under artesian conditions. Since here the water table is fairly close to the surface, the shallow aquifers are recharged by direct accretion from precipitation. Recharge to the artesian aquifers is mainly controlled by the difference in head between the water in the artesian aquifer and that in the unconfined aquifers and also by that in other artesian aquifers above and below.

The direction of ground water movement is normal to the piezometric contours and from points of higher potential to those of lower potential. While recharge can take place in the outcrop areas, field inspection indicated that the aquifers of the Middendorf Formation receive most of their recharge by leakage from overlying aquifers, and actually are discharging in the outcrop areas. Water moves down the hydraulic gradient (which may be in a different direction from the dip) to discharge into the overlying strata in areas of lower head or piezometric lows. Thus discharge is controlled both by water moving toward the piezometric low along the river and by moving down the dip finally discharging by upward leakage into the Black Creek Valley.

It is believed that the above existing conditions are similar to those in the Savannah River Valley in the vicinity of Aiken, South Carolina. There the Ground Water Branch of the USGS proved the existence of a piezometric low along the Savannah River and the discharge of artesian water into the Savannah Valley. Thus, in the Black Creek Valley (which is a piezometric low), there is a net ground water discharge rather than a net recharge to the Middendorf Formation.

George E. Siple, USGS District Geologist, Columbia, South Carolina in an article in the Journal American Water Works Association (AWWA) (Reference 2.4.7-1) states that "---the static head of water in the Middendorf Formation is about 330 ft above mean sea level at Bethune---the gradient in this part of

the coastal plain amounts to about 2.9 ft per mile." Because Lake Robinson is approximately 10 miles down dip from Bethune, the static head in the interfluvial ridges bounding Black Creek should be about 300 ft above mean sea level or about 80 ft above the maximum lake level (222 ft).

A hydrological test program will be conducted. Holes will be drilled at selected points in the vicinity of the lake to measure ground water levels, flow directions and soil permeability.

Municipal and industrial ground water usage within a 20 mile radius of Lake Robinson is obtained primarily from artesian wells. All domestic water usage in the vicinity of the plant is artesian in origin. Table 2.4.7-1 summarizes data abstracted from records on file, USGS, Water Resources Division, Ground Water Branch, Columbia, South Carolina.

Municipal and industrial sources of potable water within a 20 mile radius of the Robinson site are obtained from ground water sources. With the construction of Unit 2, two wells, of approximately 220 gpm capacity each, were provided at the Robinson site. These wells furnish water for boiler makeup for Unit 1 and for domestic (sanitary) uses for both units. This water is taken directly from the wells for such purposes as drinking fountains, lab sinks, washing of machinery, and emergency seal water at the intake structure. This water is not processed after use but sent directly to the storm drains.

Unit 2 required three wells, and a total of approximately 10,000 gpd is taken from these wells in the operation of the plant. Each well is rated at 200 gpm and in total the three are capable of providing a continuous flow of 600 gpm for 72 hr with a somewhat decreased flow indefinitely.

The wells provide makeup water for the primary and secondary systems and for specific major cooling facilities in the plant such as the component cooling loop, the residual heat removal loop and the spent fuel pit cooling loop. Water for these purposes is purified by demineralization and necessary additives are provided for effective plant operation as required. Makeup water to the primary reactor cooling system is added through the chemical and volume control system.

CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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TABLE 3.2.2-5

WASTE DISPOSAL SYSTEM CODE REQUIREMENTS

ITEM	CODE
Chemical Drain Tank	No Code
Reactor Coolant Drain Tank	ASME III,* Class C
Sump Tanks	No Code
Spent Resin Storage Tank	ASME III, Class C
Gas Decay Tanks	ASME III, Class C
Waste Holdup Tank	No Code
Water Condensate Tank	No Code
Laundry and Hot Shower Tank	No Code
Waste Evaporator	No Code
Waste Filter	ASME III, Class C
Piping and Valves	USAS-B31.1** Section 1

*ASME III-American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IV, Nuclear Vessels

**USAS-B31.1-Code for pressure piping and special nuclear cases where applicable.

The most highly stressed existing interior bearing walls, with the additional compressive force from the waste evaporator enclosure, is stressed to 850 psi. This is less than the allowable stress of 1150 psi permitted by the ACI 318-63 code. Therefore, the existing exterior and interior walls of the RAB are structurally adequate for the additional load imposed by the new waste evaporator enclosure.

3.8.4.1.2 Spent Fuel Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor. The spent fuel storage pit is constructed of reinforced concrete having 3 to 6 ft thick walls and is Class I seismic design. The entire interior basin face and transfer canal is lined with stainless steel plate.

3.8.4.1.3 Class I Section of the Turbine Building

The Class I portion of the Turbine Building is north of the Class III portion and is a separate structure. All framework and equipment supports have been designed to Class I seismic design criteria. The sum of primary stresses resulting from operating conditions and the stresses resulting from the design earthquake was limited to 133 percent of allowable stresses, as permitted by the Uniform Building Code.

All safeguards equipment in this Class I area is located on the ground floor (Elevation 226 ft). There is a Class I concrete ceiling over the top of this area to protect it from above. In addition, the Class I trench in this area is below grade with a heavy boiler plate which covers and protects the contents of the trench from falling debris in the event of an earthquake.

3.8.4.1.4 Intake Structure

The intake structure is designed as Seismic Class I, and is therefore not subject to collapse under earthquake loading. The only part of the Service Water System which is not seismic Class I design is the section in the Turbine Building.

The four service water pumps are located in three separate bays in the intake structure, the middle bay containing two pumps. The walls separating the bays and the deck above the piping are two and one half feet thick reinforced concrete.

3.8.4.1.5 Concrete Masonry Walls

In accordance with the requirements of Nuclear Regulatory Commission's IE Bulletin 80-11, the concrete masonry walls which are in the proximity of safety-related systems or equipment in the Reactor Building, Reactor Auxiliary Building, and the Fuel Handling Building have been analyzed and reinforced as necessary with structural steel supports to ensure that these walls will not collapse due to the hypothetical earthquake. Structural details are contained in References 3.8.4-2 and 3.8.4-3. The location of the concrete masonry walls is shown on Figure 3.8.4-2.

3.8.4.1.5.1 Design Criteria

The following criteria were used to analyze the concrete masonry walls and supports:

a) Material Specifications

1) Concrete block is Class B (f'c = 3,000 psi)

2) Mortar is Class S, (f'c 1,800 psi)

3) Structural steel and plate is ASTM A36

4) Concrete anchors are Phillips Wedge Anchors as manufactured by the Phillips Drilling Company, and

5) Welding electrodes conform to AWS A5.1 low hydrogen Class E70XX for Manual Shielded Metal-Arc Welding or AWS A5.17 F7X for Submerged Arc Welding.

b) Design Codes

1) American Concrete Institute (ACI) 67-23, Concrete Masonry Structures - Design and Construction

2) ACI Standard, "Building Code Requirements for Concrete Masonry Structures", (ACI 531-79)

3) American Institute of Steel Construction, (AISC) - Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings - Sixth Edition, Revised 1963

4) American Welding Society (AWS) D1.1.76 - Structural Welding Code, and

5) Phillips Catalog F-1000 dated May 1, 1973.

c) Design Loads

1) Dead Load (D)

(a) Concrete Masonry = 143 pcf

(b) Structural Steel = 490 pcf

2) Seismic Loads

(a) Hypothetical Earthquake (E) 0.2g base horizontal ground acceleration and 0.134g base vertical acceleration acting simultaneously

(b) Acceleration coefficients were obtained from the appropriate acceleration curves (Reference 3.8.4-3)

d) Load Combinations and Allowable Stresses

1) For concrete masonry

(D) + (E) using allowable stresses as per ACI 67-23

2) For structural steel

(D) + (E) using normal AISC working stresses

3.8.4.1.5.2 Analysis and Results

The block walls in question were first analyzed assuming normal ACI 531 allowable stresses in the mortar in horizontal joints as well as between wythes (collar joints) and between vertical edge of the block wall and the reinforced concrete wall or the floor to which it is joined. Based on these assumptions, walls were found to be satisfactory as they stood with no collar stress exceeding 2.5 psi as opposed to 8 psi generally allowable. Other mortar stresses were within the allowables of ACI 531.

In addition the walls were analyzed assuming only the allowable stress in the horizontal joints between blocks. No bond or friction was assumed between wythes nor between the edge of the block wall and its adjoining reinforced concrete wall. Under these assumptions additional supports were provided to ensure the walls retained their function during and after the hypothetical earthquake.

REFERENCES: SECTION 3.8 (Cont'd)

3.8.1-28 Thalmann, E. H., "Cathodic Protection in the Design of Power Plants," Pacific Coast Electrical Assoc., Joint Session of Substation and Design Group, Berkeley, California, March 31 - April 1, 1960.

3.8.1-29 Peabody, A. W., "Control of Pipeline Corrosion," p. 122.

3.8.1-30 Trouard, S. E., "Cathodic Protection of the Coated Steel Gas Main Distribution System in New Orleans," Corrosion, March 1967.

3.8.1-31 Meyer, Steigel, W. N., Waters, G. E., and Slater, J. E., "Inspection of Containment Prestressing Surveillance Tendon from H.B. Robinson Unit No. 2 Plant," Battelle Columbus Laboratories, September 28, 1976.

3.8.1-32 Meyer, Steigel, W. N., Berry, W. E., and Boyd, W. K., "Failure Analysis of a Reinforcing Rod from the Containment Prestressing Surveillance Tendon from H.B. Robinson Unit No. 2 Plant," Battelle Columbus Laboratories, March 25, 1977.

3.8.1-33 "Incidence of Corrosion on Prestressing Steel Tendons at H.B. Robinson Steam Electric Plant, 1970 - 700,000 KW Extension - Unit No. 2," CP&L/Ebasco, March 1968.

3.8.3-1 Fauske, H. K., "The Discharge of Saturated Water Through Tubes," Seventh Annual National AIche-ASME Conference, August 1964.

3.8.4-1 "Waste Evaporator Enclosure - Design and Analysis Report," Carolina Power and Light Company, H.B. Robinson Steam Electric Plant, 700,000 KW Extension - Unit 2, Ebasco Services, Inc., April 1977.

3.8.4-2 Letter, NO-80-1015, dated July 7, 1980, from B. Furr (CP&L) to J. O'Reilly (NRC) Response to IE Bulletin 80-11, H.B. Robinson Steam and Electric Plant, Unit 2.

3.8.4-3 Letter, NO-80-1633, dated November 5, 1980, from B. Furr (CP&L) to J. O'Reilly (NRC) Response to IE Bulletin 80-11 (180 Day Response), H.B. Robinson Steam and Electric Plant, Unit 2.

3.8.5-1 Hetenyi, M., "Beams on Elastic Foundation," The Univ. of Mich. Press, Ann Arbor, Mich. (1946).

3.8.5-2 Johnson, S. M., "Determining the Capacity of Bent Piles," Proc., Jour. of Research, Nat'l. Bur. of Stds., 66C, No. 3, 223-224 July-September (1962).

3.8.5-3 Rmonoff, Melvin, "Corrosion of Steel Piping in Soils," Jour. of Research, Natl. Bur. Stds, 66C, No. 3, 223-224 (1962).

The tubes were designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psig as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

When the plant was designed, no significant corrosion of the Inconel tubing was expected during the lifetime of the plant. The corrosion rate reported in Reference 3.9.3-8 shows "worst case" rates of 15.9 mg/dm^2 in the 2000 hr test under steam generator operating conditions. Conversion of this rate to a 40 year plant life gives a corrosion loss of less than 1.5×10^{-3} in., which is insignificant compared to the nominal tube wall thickness of 0.050 in.

In the case of a primary pressure loss accident, the secondary to primary pressure differential can reach 1100 psig. This pressure differential is less than the primary to secondary design pressure differential (1520 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III rules for normal operation are experienced on the tube sheet for this accident case. Actual pressure tests of 3/4 in. OD/.058 in. wall Inconel tubing show collapse under external pressure of 5700-5900 psi. Extrapolating this data to 7/8 in. OD/.050 in. wall tubes, collapse would occur at about 2630 psi at 650°F. This gives a factor of safety of 2.4 against collapse under the 1100 psig accidental application of external pressure to tubes. A check of the ASME Section VIII design curves for Iron-Chromium-Nickel Steel cylinders under external pressure indicates a predicted collapse pressure for the tubes of 2310 psi, which is close to the extrapolated value for the experimental results.

Consideration has been given to the superimposed effects of secondary side pressure loss and the maximum potential earthquake loading. The fluid dynamic forces on the internal components affecting the primary to secondary boundary (tubes) has been considered as well. For this condition, the criterion is that no rupture of primary to secondary boundary (tubes and tube sheet) occurs.

For the case of the tube sheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 10 psi (for vertical shock). Such an increase is small when compared to the pressure differentials (up to 2485 psig) for which the tube sheet is designed. Under horizontal shock loading of the maximum hypothetical earthquake, the stresses are less than those for 1.0g gravity loading experienced in a horizontal position, which the design can readily accept.

The fluid dynamic forces on the internals under secondary steam break accident conditions indicate, in the most severe case, that the tubes are adequate to constrain the motion of the baffle plates with some plastic deformation, but boundary integrity is maintained.

The ratio of the allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses is summarized in Table 3.9.3-5.

3.9.3.4 System Integrity

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification was prepared by the manufacturer. The analysis showed that the reactor vessel, steam generator, pump casing, and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the reactor coolant piping showed that it complies with the stress limits of the applicable sections of USAS B31.1.

As part of the design control on materials, Charpy V-notch toughness test curves were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator, and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, drop-weight tests were performed on the reactor vessel plate material.

As an assurance of system integrity, all components in the system were hydrotested at 3110 psig prior to initial operation.

3.9.3.5 Design and Installation Details for Mounting of Pressure-Relief Devices

In April, 1970 during hot functional testing of secondary safety valve lifting setpoints, a 6 in., Schedule 80 connection pipe to safety valve SVI-4-C failed allowing the valve to be blown completely off the connecting pipe (Reference 3.9.3-9). The failure was caused by an overload condition at a thickness change location; however, the actual failure scenario was not identified. The inlet nozzles to 12 main steamline safety valves were redesigned to increase the conservatism of the stress level and to decrease the possible stress intensification. The nozzle size was increased to an 8 in., Schedule 160 pipe. The stress calculations are shown in Reference 3.9.3-9.

REFERENCES: SECTION 3.9

3.9.2-1 Shock and Vibration Handbook, edited by Harris and Crede, Volume 3, Chapter 50: "Vibration of Structures Induced by Seismic Waves" by George W. Housner.

3.9.3-1 Letter, GD-78-076, dated January 12, 1978, E. Utley (CP&L) to R. Reid (NRC), Response on Steam Generator and Reactor Coolant Pump Support Materials.

3.9.3-2 Letter, GD-79-1861, dated July 20, 1979, E. Utley (CP&L) to J. O'Reilly (NRC), Supplemental Response to IE Bulletin 79-02.

3.9.3-3 Letter, GD-79-1719, dated July 9, 1979, E. Utley (CP&L) to J. O'Reilly (NRC), Response to IE Bulletin 79-02.

3.9.3-4 License Event Report, LER 79-037, CP&L to NRC, October 29, 1979.

3.9.3-5 License Event Report, LER 79-039, November 7, 1979.

3.9.3-6 License Event Report, LER 80-021, October 2, 1980.

3.9.3-7 Letter, dated March 22, 1982, P. Howe (CP&L) to S. Varga (NRC), In-Service Inspection and Test Program

3.9.3-8 Berry, W.E. and Fink, F.W., "The Corrosion of Inconel in High Temperature Water" Baltelle Memorial Institute, April, 1958.

3.9.3-9 Incident Report, H.B. Robinson Unit No. 2 Steam Pipe Break, NRC Docket 50261-37, June, 1970.

3.9.4-1 T. Marita, et. al., "Topical Report - Power Distribution Control and Load Following Procedures", WCAP-8403, September, 1974.

3.9.6-1 Attachment to Letter No-81-2160, dated December 31, 1981, P. Howe (CP&L) to S. Varga (NRC), In-Service Inspection and Test Program - Interval 2, March 7, 1981 to March 7, 1991.

3.9.6-2 Letter, GD-79-2186, dated September 6, 1979, B. Furr (CP&L) to J. O'Reilly (NRC), Response to IE Bulletin 79-17.

3.9.6-3 Letter, GD-79-3118, dated December 6, 1979, B. Furr (CP&L) to J. O'Reilly (NRC), Response to IE Bulletin 79-17.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The following information is taken from the original Final Safety Analysis Report (FSAR). Environmental qualifications of mechanical and electrical equipment are currently under review and are being conducted in accordance with the requirements of IE Bulletin 79-01B (Reference 3.11.0-1).

3.11.1 ENVIRONMENTAL DESIGN

The engineered safety features (ESF) instrumentation equipment inside the containment is designed to operate under the accident environment of a steam-air mixture and radiation. Electrical equipment for the ESF is located inside the containment, in the Auxiliary Building and turbine-generator structure. The equipment located inside the containment which must function in the post-accident environment and their expected required operational time is given below:

a) Safety Injection System (SIS) and Containment Spray System actuation sensors (first five minutes after accident)

b) SIS motor operated valves and flow instrumentation (first five minutes after accident)

c) Accumulator level instrumentation (first five minutes after accident)

d) Containment fan coolers (three hours)

e) Containment sump level instrumentation (three hours)

f) Air and motor operated containment isolation valves (operation completed in first five minutes after accident), and

g) Power and instrumentation cables for the above listed equipment.

The reactor protection control and instrumentation equipment and electrical equipment for ESF located in the Auxiliary Building and turbine-generator structure will operate in a normal ambient environment following a major loss of coolant accident (LOCA). Auxiliary Building equipment in the containment sump water recirculation loop, required during the recirculation mode, is listed below:

a) Residual heat removal (RHR) pumps

b) Flow, temperature and pressure instrumentation for the Residual Heat Removal System (RHRS), and

c) Power and instrument cables for pumps.

The design considerations and specifications to be used in the selection of motors which must function in the post-accident environment are discussed in Sections 6.1 through 6.3. Similar application criteria apply to the specifications of control and instrumentation equipment and other electrical equipment.

Areas of high radiation would exist inside the containment and in those portions of the Auxiliary Building near RHRS equipment following a major LOCA. The maximum dose rates within the containment would be approximately 4.2×10^6 rad per hour or 2.6×10^7 rad per week. The maximum dose rates in high radiation areas of the Auxiliary Building (RHR compartments) would be less than one percent as high. The ability of electrical equipment in the emergency core cooling system (ECCS) to withstand radiation exposure would be limited by radiation effects on electrical insulation materials and motor bearing lubrication.

The electrical equipment for the ECCS located in the containment use only inorganic, silicone, and epoxy plastic insulating materials. These materials have a threshold for radiation damage which might affect their function of 10^8 rad or higher. They would therefore, provide considerable margin above the maximum post-accident radiation dose that would result from the exposure times specified earlier. The lower ambient temperatures and radiation levels in the Auxiliary Building will permit the use of normal elastomer or plastic insulation materials. These materials have a threshold for radiation damage of 10^6 rad or higher.

Where required because of a location in possible high radiation areas, motor bearings will be lubricated with radiation-rated lubricants.

CHAPTER 4
REACTOR

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TABLE 4.1.2-2

THERMAL-HYDRAULIC DESIGN VALUES

Rated Heat Output, Mwt	2300
Maximum Overpower, %	12
Heat Generated in Fuel, %	97.4
Nominal Design Pressure, psia	2250
Nominal Inlet Temperature, ^{o}F	551.9
Average Core Temperature, ^{o}F	577.8
Nominal Outlet Temperature at Hot Channel, ^{o}F	637
Total Reactor Coolant Flow, lb/hr	101.5×10^6
Active Coolant Flow, lb/hr	97.0×10^6
Average Mass Velocity, lb/hr	2.34×10^6
Average Coolant Velocity Along Fuel Rods, ft/sec	15.0
Active Heat Transfer Surface Area, ft^2	42,662
Average Heat Flux, Btu/hr-ft^2	179,218
Maximum Heat Flux, Btu/hr-ft^2	469,551
Maximum LHGR, kW/ft	15.27
Average LHGR, kW/ft	5.83
Core Pressure Drop, psi*	$19.7 \pm .8$

*Includes static head.

4.4.3 INSTRUMENTATION REQUIREMENTS

The following sections describe the instrumentation requirements for the reactor core. Chapter 5 of the Updated FSAR contains a description of the instrumentation requirements for the RCS.

4.4.3.1 Incore Instrumentation

The incore instrumentation system consists of 51 thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected locations; and 50 flux thimbles, which run the length of selected fuel assemblies for measurement of the neutron flux distribution within the core. Five movable miniature neutron flux detectors with associated control and readout equipment may be used to scan the length of 46 selected fuel assemblies to provide remote reading of the axial flux distribution. Four flux thimbles located symetrically around the core contain fixed incore neutron flux detectors (four detectors per thimble). These fixed incore detectors with associated readout equiment provide continuous monitoring of neutron flux within the core. The incore instrumentation system is shown in Figure 4.4.3-1.

The experimental data obtained from the incore temperature and flux distribution instrumentation system, in conjunction with previously determined analytical information, can be used to determine the fission power distribution in the core at any time throughout core life. Once the fission power distribution has been established, the maximum power output is primarily determined by thermal power distribution and the thermal and hydraulic limitations which determine the maximum core capability.

The incore instrumentation provides information which may be used to calculate the coolant enthalpy distribution, the fuel burnup distribution, and to estimate the coolant flow distribution.

4.4.3.2 Overtemperature and Overpower ΔT Instrumentation

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 112 percent of design power density and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification and the increase in rated thermal output to 2300 Mwt on core safety limits. The revised setpoints in the Technical Specifications ensure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figures 4.4.3-2 and 4.4.3-3.

4.4.3.3 Instrumentation to Limit Maximum Power Output

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.30 based on steady state nominal operating power

levels less than or equal to 100 percent, steady state nominal operating RCS average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2 percent in power, +4°F in RCS average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 45 percent.

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

LIST OF TABLES

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

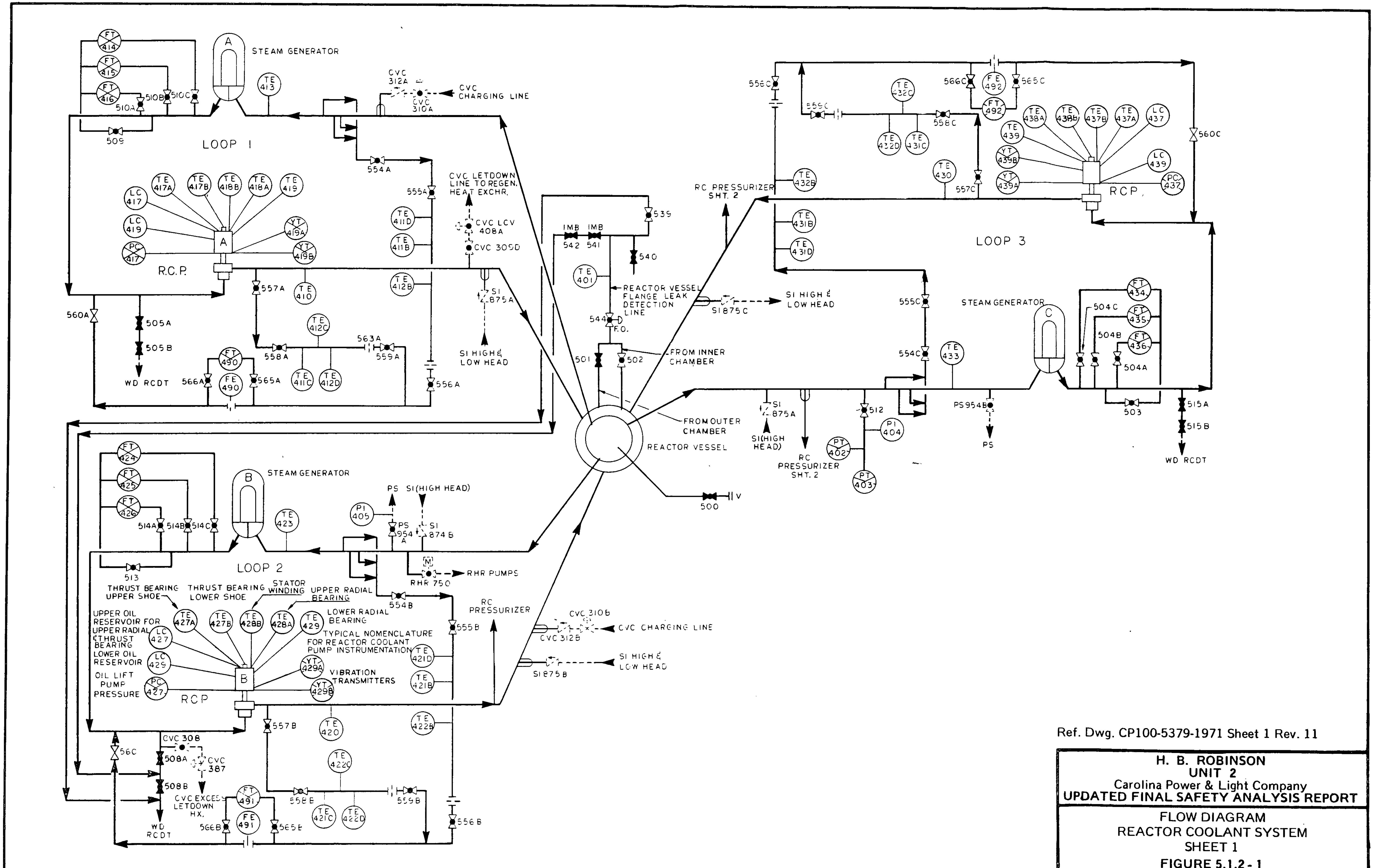
LIST OF FIGURES

TABLE 5.1.0-1

REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

Total Primary Heat Output, Mwt	2308
Total Primary Heat Output, Btu/hr	7875×10^6
Number of Loops	3
Coolant Volume (liquid), Including Pressurizer Volume, ft^3	9343
Total Reactor Coolant Flow, gpm	265,500

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valves (open)	2260
High Pressure Trip	2385
High Pressure Alarm	2335
Low Pressure Trip	1700
Low Pressure Alarm	2135
Hydrostatic Test Pressure	3110



Ref. Dwg. CP100-5379-1971 Sheet 1 Rev. 11

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM
REACTOR COOLANT SYSTEM
SHEET 1

FIGURE 5.1.2 - 1

5.2.2 OVERPRESSURIZATION PROTECTION

The Reactor Coolant System (RCS) is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. These power-operated relief valves and code safety valves are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray.

5.2.2.1 Design Bases

The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 percent, in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. The capacity of the pressurizer safety valves is determined from considerations of the reactor protective system and accident or transient conditions which may potentially cause overpressure.

Details of the analysis are reported in Chapter 15.0. Experience has shown that the safety valve capacity so determined is adequate for all the other transients, as the results of Chapter 15.0 show.

5.2.2.2 Design Evaluation

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting. The pressurizer safety and relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 5.1.2-2, and the valve design parameters are given in Table 5.4.10-1. Valve sizes are determined as indicated above.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure value by rupture discs which discharge into the reactor containment.

5.2.2.3 Operation Below 350°F

The pressurizer power operated relief valves (PORV) are utilized to protect against exceeding safe pressure limits under low temperature conditions (Reference 5.2.2-1). A manual permissive switch is utilized to arm overpressure protection channel (one for each PORV) when the plant is below 350°F and capable of a solid water condition (no steam bubble in the pressurizer). Operating solid can produce extreme pressure spikes that are not encountered when there is the cushioning effect present from the steam bubble during normal plant operation. A nonredundant temperature comparator provides an annunciator signal when the loop temperature drops below 350°F.

During normal plant operation, the permissive switch is not armed and the PORV are not operable in the low temperature/overpressure mode. With the system not armed, a reduction in temperature below 350°F (normal cooldown procedures or abnormal conditions) causes the annunciator to energize signaling the operator to arm the overpressure protection system via the permissive switch. Redundancy for the arming function is provided by the plant operating procedures which require arming the system before the temperature drops below 350°F. When armed, the PORV become operable. Exceeding the setpoint with the system armed causes the PORV to open. The annunciator does not light under normal conditions if the system was armed prior to the temperature dropping below 350°F. With the system armed below 350°F, the next energizing of the annunciator indicates the setpoint has been exceeded and the PORV will open.

5.2.3 REACTOR COOLANT BOUNDARY MATERIALS

5.2.3.1 Material Specifications

Each of the materials used in the RCS is selected for the expected environment and service conditions. The major component materials are listed in Table 5.2.3-1.

5.2.3.2 Compatibility with Reactor Coolant

All reactor coolant system materials which are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 5.2.3-2. The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces.

All materials exposed to reactor coolant are corrosion resistant. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water quality listed in Table 5.2.3-2. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System (CVCS) and Sampling System which are described in Section 9.3.4.

5.2.3.3 Fabrication and Processing of Ferritic and Austenitic Stainless Steel Materials

Table 5.2.4-1 summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 5.2.4-1, there are those which the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication were governed by United States American Standards (USAS) B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified Westinghouse engineers. These procedures were developed to provide the highest assurance of quality material and fabrication. They considered not only the size of flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, received a 100 percent surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed. All reactor coolant plate material was subjected to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. (All forgings received the same inspection.) In addition, 100 percent of the material volume was covered in these tests as an added assurance over the grid basis required in the code.

Field erection and field welding of the reactor coolant system were performed such as to permit exact fit up of the 31 in. inside diameter (ID) closure pipe subassemblies between the steam generator and the reactor coolant pump (RCP). After installation of the pump casing and the steam generator, measurements were taken of the pipe length required to close the loop. Based on these measurements, the 31 in. ID closure pipe subassembly was properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle.

The phenomena of stress corrosion cracking and corrosion fatigue are not encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time. It is characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides and free oxygen. Another environment was identified by IE Bulletin 79-17, July 26, 1979, as stagnant borated water. The program at HBR 2 developed in response to this bulletin is discussed in Section 3.9.6.

Experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment, stress corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel Alloy has excellent resistance to general and pitting type corrosion in severe operating water conditions.

The selection of Inconel as the tube material for the HBR 2 steam generator was based on considerable experience with Inconel in steam generator and heat exchanger applications. Since 1962, widespread adoption of Inconel for steam generator tubes in nuclear stations was evident: as for example, Connecticut-Yankee; San Onofre; PM-1, Sundance; PM-3A, McMurdo Sound; Carolina-Virginia Tube Reactor; NPD; and Hanford N-Reactor. Materials with lead traces in the overall composition were present in the secondary side of the referenced plants. The use of lead in the materials of the secondary side of this plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys, and is insignificant.

All external insulation of RCS components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is insulated with low halide content insulating material. All other external corrosion resistant surfaces in the RCS are insulated with low or halide-free insulating material as required.

Cleaning of RCS piping and equipment was accomplished before and/or during erection of various equipment. Stainless steel piping was cleaned in sections as specific portions of the systems were erected. Pipe and units large enough to permit entry by personnel were cleaned by locally applying approved solvents (Stoddard solvent, acetone and alcohol), and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove all trapped foreign particles.

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Material Specifications

The materials of construction of the reactor vessel are given in Table 5.2.3-1.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The reactor vessel material is heat treated specifically to obtain good notch ductility which ensures a low nil-ductility transition temperature (NDTT), and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDTT. An initial maximum value of NDTT of 40°F in this region had been established during fabrication.

5.3.1.3 Special Methods for Nondestructive Examination

Westinghouse required, as part of its reactor vessel specification, that certain special tests which are not specified by the applicable codes be performed. These tests are listed below:

a) Ultrasonic Testing - Westinghouse requires that a 100 percent volumetric ultrasonic test of reactor vessel plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid. The 100 percent volumetric ultrasonic test was a severe requirement, but it assured that the plate was of the highest quality.

b) Radiation Surveillance Program (discussed in Section 5.3.1.6)

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

The HBR 2 reactor vessel was manufactured by Combustion Engineering and shipped to the site on July 12, 1968. It was fabricated from SA302B and SA302A plate material, with three vertical weld seams in each shell course, and three shell courses, as shown in Figure 5.3.1-1. The numerical designations of the plates and welds used in the vessel are also shown in Figure 5.3.1-1, along with the location of the reactor core, relative to these materials. End-of-life fluences are identified with each weld location.

The chemistry of the three plates used on the intermediate shell course was reported in Reference 5.3.1-1 and is summarized in Table 5.3.1-1. The nickel content of the base metal, while not directly measured, is less than 0.20 weight percent, because the steel used in SA302B.

There are no records of "as deposited" weld chemistry available for the longitudinal welds in the intermediate shell, but a representative chemistry is reported in Reference 5.3.1-1 and repeated in Table 5.3.1-1. The specific weld tested was representative of the intermediate to nozzle shell circumferential seam and is identified as a high nickel weld.

All the longitudinal welds were made at Combustion Engineering during a time period when low nickel welds were the standard practice. Welds during this time period were made with RACO 3 weld wire and ARCOS B-5 flux. A series of chemistry analyses from welds made for other vessels at Combustion Engineering during this time frame is provided in Table 5.3.1-2. This table provides representative properties for this type of weld, from which can be inferred the properties of the HBR intermediate shell welds.

A study was made of the available weld inspection records from Combustion Engineering to identify the date at which the practice of adding nickel to the welds was begun. When nickel was added, an additional weld wire was added to the RACO 3, and this was called out as the weld inspection records as "Ni 200". A summary of available weld inspection records is provided in Table 5.3.1-3, and from the table it can be seen that the addition of nickel began sometime between April, 1965 and September, 1966. Weld seams identified for plant B on Table 5.3.1-3 are for the HBR 2 reactor vessel.

Identification of the beltline region welds are presented in Table 5.3.1-4. This information was obtained from weld inspection records from Combustion Engineering. As noted in this table, the circumferential seams were welded using the practice of adding nickel to the weld. When the nickel was added an additional weld wire was called out on the Combustion Engineering reports as shown on Table 5.3.1-4 as "Ni 200". As seen from Table 5.3.1-3, longitudinal weld seams were made in 1964 prior to the practice of adding nickel to welds so that nickel content of the welds is low, < .20 percent.

Because both high nickel and low nickel welds were used in the beltline region, both trend curves derived for low nickel materials and Regulatory Guide 1.99 Rev. 1 were used in the analyses. The technical basis for the low nickel trend curves is provided in WCAP 10019 (Reference 5.3.1-2).

The weld seams in the Robinson vessel were constructed with backchipping at the inner surface of the vessel which removed material from the inner surface to a stated depth of 3/8 in. for the circumferential weld. The backchipped regions were subsequently rewelded so that the weld in this region contains no more than 0.05 percent Copper. The justification of this weld chemistry is given in Reference 5.3.1-3. This effect is used as a secondary argument for circumferential weld integrity. Backchipping is not considered for the longitudinal welds since the inner surface was machined, thereby removing the backchipped region.

5.3.1.5 Fracture Toughness

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as American Society for Testing and Materials (ASTM)-A302 Grade B parent material of the H. B. Robinson Unit 2 (HBR 2) reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility under certain conditions of irradiation. In pressure vessel material, the most serious mechanical property change is the reduction in the upper shelf impact strength. Accompanying the decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

The method for guarding against fast fracture in reactor pressure vessels presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of:

a) The drop weight nil-ductility transition temperature (NDTT per ASTM E-208)

b) The temperature 60^{o}F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material.

However, the identification of a value of RT_{NDT} which could limit further safe operation of a reactor pressure vessel is not appropriate. Although such a simplistic criterion is perhaps desirable, it should not be used as the sole parameter to determine the acceptability of the reactor vessel for any specific plant, or to compare plants. The acceptability of a vessel for continued operation is dependent on many variables in addition to the material properties, and therefore a limiting material property could only be established for a prescribed transient with a fixed set of conditions. The most appropriate criteria for continued operation should be based on fracture analysis results, since these results incorporate all the variables which can affect integrity. The criteria adopted have been detailed in WCAP-10019 (Reference 5.3.1-2).

The value of RT_{NDT} at the end of vessel design life was calculated using the trend curves for low nickel materials for the longitudinal welds and the Reg. Guide 1.99, Rev. I trend curve for the circumferential welds and plate material. These values are shown in Table 5.3.1-5 as well as the RT_{NDT} at the inner surface and at the quarter thickness location. The values of RT_{NDT} indicate that the circumferential weld would be limiting but the stresses could be higher in the longitudinal seams, so that both were evaluated in the analyses for HBR 2. Additional plant specific analyses (Reference 5.3.1-4) have been performed to supplement and supersede some of the results reported in Reference 5.3.1-2.

All the limiting thermal transients were evaluated on a plant specific basis to evaluate the impact of key plant specific parameters such as fluence, material properties, vessel geometry and weld locations on the acceptable vessel lifetimes for HBR 2.

Based on these analyses (Reference 5.3.1-2 and 5.3.1-4), the achievement of end of design life for the vessel, as a minimum, would be expected for all the limiting transients with the benefit of warm-prestressing. The maximum value of RT_{NDT} at end-of-life is not suggested to be a limiting RT_{NDT} value for HBR 2. Violation of acceptance criteria described in Reference 5.3.1-2 would not be expected to occur until some time after end-of-life. In summary, the most appropriate criteria for continued operation should be based upon fracture analysis results. These results incorporate all the variables affecting vessel integrity.

Reference 5.3.1-2 includes the analytical results of the most limiting transients with regard to vessel thermal shock consideration for HBR 2. Table 5.3.1-6 summarizes the minimum number of additional years of vessel operation without violation of acceptance criteria for the transients that were evaluated and provides additional information with respect to fundamental inputs to each analysis.

Detailed integrity assessments have been carried out for the HBR 2 reactor vessel, postulating the occurrence of five different types of thermal shock events:

- o Large Loss of Coolant
- o Small Loss of Coolant
- o Large Steam Break
- o Small Steam Break
- o Rancho-Seco Transient

A complete discussion of the transients and the basis for the thermal, stress and fracture analyses which were performed is provided in Reference 5.3.1-2. Included in Table 5.3.1-6 are two new analyses, for the Small Steam Break and Rancho Seco transients, using plant specific property information which are more recent than those done in Reference 5.3.1-2. These results are considered to be more appropriate to the HBR vessel, and therefore should replace the previously obtained results.

The results obtained on all five of the transients showed that vessel integrity would be maintained throughout the design lifetime of the plant. To reach this conclusion the fracture analyses were evaluated against the acceptability criteria described below.

The results of the fracture mechanics analysis of postulated longitudinal and circumferential flaws are presented in terms of the maximum number of calendar years the reactor vessel will conform to the following criteria:

a) Minimum critical flaw depth for crack initiation is greater than 1.0 in., or

b) Crack arrest occurs within 75 percent of the vessel wall thickness.

The initiation criterion is based on the ultrasonic inspection limitations, and the arrest criterion is set to be consistent with Appendix A of Section XI, ASME Code.

It should be noted that the acceptable vessel lifetimes given in Reference 5.3.1-2 are based on this acceptance criteria, and therefore, the defined acceptable lifetime does not indicate catastrophic failure of the vessel.

The results obtained for the HBR 2 vessel made use of the principal of warm prestressing to demonstrate integrity for the remaining design lifetime for the limiting transients presented.

The technical basis for the use of warm prestressing in demonstrating vessel integrity has been given in detail in Reference 5.3.1-2. The application of warm prestressing to the transients results in an excellent behavior with regard to vessel integrity, because warm prestressing occurs very early in each transient.

The specific results for the two inch, no mixing, small loss-of-coolant accident the most limiting transient, are provided in Figure 5.3.1-2 for the longitudinal low nickel weld.

The curves for the longitudinal low nickel weld are shown since it is subjected to the highest stresses. As can be seen from the figure, the crack will arrest at 1024 seconds into the transient based on the warm-prestressing effects as discussed in Reference 5.3.1-2.

Reference 5.3.1-2 gives the benefit obtained by installation of a low leakage core. The HBR 2 reactor vessel will have a low leakage core installed in 1982. The expected benefit is to reduce the core leakage flux and thus reduce the radiation damage to the reactor vessel. For this particular cycle, the low leakage core also provides desirable characteristics from neutron physics and economic viewpoints. Future cores will be evaluated based on the same characteristics.

Detailed analyses have been carried out for bounding postulated transients which result in thermal shock to the reactor pressure vessel. The results of these analyses are summarized in Table 5.3.1-6 and show that the HBR 2 plant can continue operation through end-of-life before the reactor vessel integrity acceptance criteria could be violated.

5.3.1.6 Material Surveillance

In the reactor vessel surveillance program, the evaluation of radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) test specimens. A description of the program basis including the material to be tested, specimen, and capsule design, and pre-irradiation test results is presented in Reference 5.3.1-5. This program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

The reactor vessel surveillance programs use eight specimen capsules which are located about 3 in. from the vessel wall directly opposite the center portion of the core (see Figure 5.3.1-5). The capsules contain reactor vessel steel specimens from the shell plates or forgings located in the core region of the reactor and associated weld metal and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee E10 Radioisotopes and Radiation Effects are inserted in the capsules. The eight capsules will contain at least 27 tensile specimens, 288 Charpy V-notch specimens (which will include weld metal and heat affected zone material) and 36 WOL specimens. Dosimeters including pure Ni, Al-Co, (0.15 percent), Cd shielded Al-Co, Cd shielded Np-237 and Cd shielded U-238 are placed in impact specimens, tensile specimens or filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion.

The analytical methodology and the design basis currently used to predict time averaged fast neutron flux and fluence levels within the pressure vessel/surveillance capsule geometry are discussed in detail in Reference 5.3.1-2.

Additional HBR 2 plant specific analyses have been performed to include geometric, material, and power distribution information fully consistent with the above methodology and to provide a sound basis for the prediction of the long term fast neutron environment to which the pressure vessel will be exposed (Reference 5.3.1-4).

Also included is a summary of the results of the latest design basis neutron transport calculation performed for this vessel as well as an updated evaluation of neutron dosimetry from each of the reactor vessel surveillance capsules which have been withdrawn to date. This dosimetry re-evaluation not only reflects advances in dosimetry analysis methodology and nuclear data, but in addition establishes dosimetry results for all capsules on a consistent basis suitable for direct comparison with analytical predictions.

Geometric information for use in neutron transport calculations is provided in Figures 5.3.1-3 through 5.3.1-5. In Figure 5.3.1-3, a plan view of the reactor at the core midplane is depicted. This figure shows the reactor core, lower internals, pressure vessel, and the inner diameter of the primary biological

shield. Pertinent dimensional information is also included in Figure 5.3.1-3. In Figure 5.3.1-4, a detailed description of the surveillance capsule geometry and associated structure is provided. This information is sufficient to allow accurate determinations of capsule lead factors as well as spectrum averaged reaction cross-sections for dosimetry applications. In Figure 5.3.1-5, the azimuthal location of each of the capsules included in the reactor vessel surveillance program is illustrated.

Since initial startup when the surveillance program was established based on Reference 5.3.1-5, two surveillance capsules have been withdrawn from the HBR reactor. In 1973, Capsule S was removed from the 10^{o} azimuthal position while in the 1975-1976 outage, Capsule V was withdrawn from the 20^{o} location. Neutron dosimetry from both capsules S and V were evaluated by Southwest Research Institute (SWRI), and the results were documented in SWRI 02-3574 and SWRI 02-4397, respectively. Early in 1977, the results for Capsule V (Reference 5.3.1-1) were revised in a letter from E. B. Norris to T. Clements (Reference 5.3.1-6).

The SWRI radiometric counting data have been extracted from the appropriate reports and the fluence determinations have been updated to reflect the following changes in surveillance dosimetry methodology.

a) Application of the best available nuclear data

b) Use of spectrum averaged cross-sections which include capsule perturbation effects

c) Spatial gradient corrections to measured count rates to permit neutron flux evaluations at the geometric center of the capsule

d) Review of standard used in Capsule V dosimetry

Updated results based on the Fe^{54} (n,p) Mn^{54} reaction are provided in Tables 5.3.1-8 and 5.3.1-9 for Capsule S and V, respectively. The Capsule S data exhibits a 7 percent variation between calculation and measurement and therefore, are supportive of the analytical neutron flux predictions. However, the Capsule V measurement exceeds prediction by some 22 percent. A discrepancy of this magnitude is slightly above that which is typical for Westinghouse reactors.

The material descriptions for each of the major zones shown in Figure 5.3.1-3 are listed in Table 5.3.1-7. The data are presented in terms of volume fractions of solid material in the defined zone of interest. Since neutron transport computations for fluence determinations are of the fixed source variety, fuel enrichment is of no consequence. However, for consistency the UO_2 material listed in Table 5.3.1-7 is taken to contain a nominal 3.2 weight percent U-235.

The core power distributions for use in the computations of time averaged neutron flux and long term neutron fluence levels are given in Figures 5.3.1-6 and 5.3.1-7 and in Reference 5.3.1-2. In Figure 5.3.1-6 the relative fuel assembly power levels are given for one core octant. The information is presented relative to a core average of 1.0. Also presented in Figure 5.3.1-6 are a series of fuel assembly numbers which are used to relate spatial power distribution gradients listed in Reference 5.3.1-2 with core location. All

fuel assemblies labeled Number 1 are assumed to have a flat power distribution; that is, no spatial gradients exist within these assemblies. Spatial gradients for assembly types 2 through 9 are tablulated in Reference 5.3.1-2. The data in Reference 5.3.1-2 is oriented such that the power value in the upper left hand corner of the table represents the portion of the fuel assembly that is closest to the center of the reactor core. Values of these spatial gradients are uniformly spaced within each fuel assembly. The time averaged axial power distribution for use in neutron transport calculations is shown graphically in Figure 5.3.1-7. As discussed in Reference 5.3.1-2, these design basis power distributions are statistically based and have proven to yield satisfactory results for long term fluence predictions.

Results of neutron transport calculations for the geometry shown in Figure 5.3.1-3 are presented in Figures 5.3.1-8 through 5.3.1-10. In Figure 5.3.1-8, calculated maximum neutron flux levels at the surveillance capsule centerline, pressure vessel inner radius, 1/4 thickness location, and 3/4 thickness location are presented as a function of azimuthal angle. In Figure 5.3.1-9, the radial distribution of maximum fast neutron flux ($E > 1.0$ Mev) through the thickness of the pressure vessel is shown. The relative axial variation of neutron flux within the vessel is given in Figure 5.3.1-10.

The data given in Figures 5.3.1-8 through 5.3.1-10 can be used directly to develop lead factors relating each surveillance capsule to any point in the pressure vessel; or, in conjunction with appropriate full power operating times, or derive fast neutron fluence distributions within the vessel. For example, critical weld locations for the reactor vessel are shown schematically in Figure 5.3.1-1. Using the flux distributions given in Figures 5.3.1-8 through 5.3.1-10, the neutron radiation levels and hence the materials properties at these weld locations can be determined for any time in plant life.

Since the longitudinal weld metal copper content is unknown, the copper content was assumed to be 0.35 weight percent, and the upper trend curve for low nickel (Reference 5.3.1-2) was used to predict irradiation effects for the intermediate shell longitudinal welds. Note that the highest copper content observed in any of the low nickel welds for which chemistry is available was 0.27 weight percent, so the use of the upper trend curve may be overly conservative.

Predicted values of RT_{NDT} for the most critical intermediate shell plate and weld materials are found in Table 5.3.1-5. Some portion of the critical shell plates is located in the peak fluence region, so this fluence was used in predictions. The governing longitudinal weld (2-273C) is that located nearest to a peak fluence region, only 4.5 degrees from the cardinal axis (Figure 5.3.1-1). The intermediate to lower shell circumferential weld was also evaluated because of its assumed high copper and known high nickel content.

The initial valves for RT_{NDT} for the intermediate shell plate and weld seams were estimated, using the NRC Branch Technical Position MTEB 5-2 (Rev. 1).

The values of RT_{NDT} after 7.2 effective full power years shown in Table 5.3.1-5 were calculated using the irradiation damage curves in Reference 5.3.1-2 for low nickel and Regulatory Guide 1.99 Rev. 1 where applicable; as was the future rate of RT_{NDT} increase listed at the bottom of the table. Note that the use of the low nickel curves result in lower predicted values of RT_{NDT} than those values

provided in earlier submittals, but the technical basis used in the predictions is more consistent with available surveillance data. The damage predictions of Table 5.3.1-5 show that the circumferential weld seam is governing. However, since the longitudinal welds are subjected to higher stress, both circumferential and longitudinal welds were considered in the analyses.

The surveilance weld for HBR 2 is representative of the nozzle shell to the intermediate shell weld circumferential seam. This weld which has a chemistry of .34 percent Cu, 0.021 percent P, and 1.20 percent Ni was found to be below the .35 percent Cu Reg. Guide 1.99 curve as shown in Figure 5.3.1-11 (RT_{NDT} = 210°F for a fluence of .508 x 10^{19} n/cm^2, Reference 5.3.1-5). However, prediction of the shift in RT_{NDT} for the circumferential welds reported in Table 5.3.1-5 is based on the .35 percent Cu Reg. Guide 1.99 curve, so that the reported RT_{NDT} shift is expected to be larger than actual for early reactor life, but would be the same for end-of-life fluence, which is based on the Reg. Guide 1.99 upper limit.

5.3.1.7 Reactor Vessel Fasteners

The reactor closure head and the reactor vessel flange are joined by 50 - 7 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leakoff connection is also provided beyond the outer O-ring seal.

The vessel closure contains fifty, 7 in. diameter studs. The stud material is ASTM A-540, which at design temperature has a minimum yield strength of 104,400 psi in accordance with Code Case 1335.2. The membrane stress in the studs when they are at the steady state operational condition is approximately 37,500 psi. This means that as few as nineteen of the fifty studs are required in order to withstand the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

TABLE 5.3.1-3

CHRONOLOGY OF WELD INSPECTION RECORDS FROM COMBUSTION ENGINEERING

WELD WIRE/FLUX	DATE	WELD LOCATION	PLANT*
RACO3, ARCOS B-5	Nov-Dec 1963	Intermed. shell vert. seams	A
"	Feb-Mar 1964	Lower shell vert. seams	A
"	Feb-Mar 1964	Lower shell vert. seams	B
"	Oct 1964	Center shell vert. seams	B
"	Oct-Nov 1964	Nozzle shell to intermed. shell	A
"	Jan-Feb 1965	Bottom head to intermed. shell	A
"	Mar-Apr 1965	Intermed. to lower shell	A
RACO 3, ARCOS B-5	Sept-Oct 1966	Nozzle shell to intermed. shell	B
+ Ni 200 Wire	June-July 1967	Intermed. shell to lower shell	B

*Note: Plant B is H. B. Robinson Unit 2

TABLE 5.3.1-4

IDENTIFICATION OF H. B. ROBINSON 2 REACTOR VESSEL BELTLINE REGION WELD METAL

WELD LOCATION	WELD PROCESS	WELD WIRE TYPE	WELD WIRE HEAT NO.	FLUX TYPE	FLUX LOT NO.
Nozzle Shell to Inter. Shell Circle Seam 10-273	Submerged Arc	RACO 3 +Ni 200	W5214 N7753A	Linde 1092	3617
Inter. Shell Vertical Seams	Submerged Arc	RACO 3	86054B	ARCOS B5	4E5F
Inter. Shell to Lower Shell Circle Seam 11-273	Submerged Arc	RACO 3 +Ni 200	34B009 N9879A	Linde 1092	3724
Lower Shell Vertical Seams	Submerged Arc	RACO 3	86054B	ARCOS B5	
Surveillance Weld	Submerged Arc	RACO 3 +Ni 200	W5214 N7753A	Linde 1092	3617

TABLE 5.3.1-5

H. B. ROBINSON UNIT 2 REACTOR VESSEL

COMPONENT	PLATE OR SEAM NO.	Cu (%)	P (%)	Ni (%)	INITIAL RT_{NDT}[a] (°F)	7.2 EFPY OPERATION: INNER SURFACE FLUENCE 10^{19} n/cm^2	INNER SURFACE RT_{NDT} (°F)	1/4 THICKNESS FLUENCE 10^{18} n/cm^2	1/4 THICKNESS RT_{NDT} (°F)
Nozzle shell to inter. shell weld	10-273	.34	.021	1.20	0	.562	247[b]	3.39	192[b]
Inter. shell plate	W10201-4	.12	.007	NA	46	1.42	141[b]	8.55	120[b]
Inter. shell long. weld	2-273C	.35	-	<.20	0	1.30	183[d]	7.84	171[d]
Inter. shell to lower shell weld	11-273	.35[c]	-	>.50	0	1.24	290[b]	7.46	261[b]
Lower shell long. weld	3-273A	.35	-	<.20	0	1.24	182[d]	7.46	170[d]

FUTURE RATE OF RT_{NDT} INCREASE[b]

Shell plant ~ 107°F for remaining design life.

Weld seams	~ °F/EFPY for next 10 EFPY	~ °F for remaining design life.
Longitudinal[d]	~ 2 to 3	~ 1 to 2
Circumferential[b]	~ 5 to 6	~ 3 to 4

(a) 1/4 thickness for shell plate and inner surface for weld seams.

(b) Based on SLOPE of prediction curves presented in Reg. Guide 1.99 Rev. 1.

(c) Estimated to be similar to weld seam 10-273 (Cu assumed to be .35% based on Reg. Guide 1.99, Rev. 1).

(d) Low Nickel Trend Curves as Presented in Reference 5.3.1-2.

TABLE 5.3.1-6

H. B. ROBINSON 2 MINIMUM NUMBER OF ADDITIONAL YEARS OF VESSEL OPERATION

WITHOUT VIOLATION OF ACCEPTANCE CRITERIA FOR THERMAL SHOCK TRANSIENTS

TRANSIENT	MINIMUM NUMBER OF ADD'L YEARS*	REMARKS						
		VESSEL GEOMETRY	WELD LOCATION	MATERIAL PROPERTIES	TRANSIENT CHARACTERISTIC	FLUENCE PROFILE	TREND CURVE	BENEFIT OF WARM-PRESTRESS
Large LOCA	>31	Plant Specific	Plant Specific	Plant Specific as given in Table	Plant Specific	Plant Specific	R.G. 1.99 Low Ni**	YES
Small LOCA	>31	Plant Specific	Plant Specific	Plant Specific as given in Table	Limiting Generic 3 Loop 2" Break - No Mixing Case	Plant Specific	R.G. 1.99 Low Ni**	YES
Large Steam Break	>31	Plant Specific	Plant Specific	Plant Specific	Plant Specific	Plant Specific	R.G. 1.99 Low Ni**	YES
Small Steam Break	>31	Plant Specific	Plant Specific	Plant Specific	Generic	Plant Specific	R.G. 1.99 Low Ni**	YES
Rancho-Seco Transient	>31	Plant Specific	Plant Specific	Plant Specific	Generic	Plant Specific	R.G. 1.99 Low Ni**	YES

* Accumulated EFPY as of 10/31/81 is 7.09 years. The values shown reflect the number of years before conservative acceptance criteria (Reference 5.3.1-2) are exceeded (does not indicate actual vessel failure) with the use of a 0.8 plant usage factor (capacity factor).

**Benefit of low nickel for longitudinal welds only.

TABLE 5.3.1-7

MATERIAL DESCRIPTION FOR USE IN NEUTRON TRANSPORT CALCULATIONS

ZONE	MATERIAL	VOLUME FRACTION
Reactor Core	Water	0.58864
	UO_2	0.29967
	Zirc - 4	0.10035
	Inconel - 718	0.00281
	Stainless Steel - 304	0.00062
Core Baffle	Stainless Steel - 304	1.0
Core Barrel	Stainless Steel - 304	1.0
Thermal Shield	Stainless Steel - 304	1.0
Surveillance Capsule Structure	Stainless Steel - 304	1.0
Surveillance Specimens	Low Alloy Steel	1.0
Pressure Vessel	Low Alloy Steel (Stainless Clad)	1.0

TABLE 5.3.1-8

COMPARISON OF MEASURED AND CALCULATED FAST NEUTRON FLUX MONITOR SATURATED ACTIVITIES FOR CAPSULE S

REACTION AND AXIAL LOCATION	RADIAL LOCATION (cm)	SATURATED ACTIVITY (DPS/mg)	ADJUSTED SATURATED ACTIVITY (DPS/mg)	FAST NEUTRON FLUX (n/cm^2-sec) CAPSULE S	CALCULATED
$Fe^{54}(n,p)Mn^{54}$					
Top	192.46	4.57×10^3	5.43×10^3	1.24×10^{11}	
Top middle	192.46	4.74×10^3	5.64×10^3	1.29×10^{11}	
Middle	192.46	4.59×10^3	5.46×10^3	1.25×10^{11}	
Bottom middle	192.46	5.26×10^3	6.26×10^3	1.43×10^{11}	
Bottom	192.46	4.40×10^3	5.24×10^3	1.20×10^{11}	
Average				1.28×10^{11}	1.19×10^{11}

TABLE 5.3.1-9

COMPARISON OF MEASURED AND CALCULATED FAST NEUTRON FLUX MONITOR SATURATED ACTIVITIES FOR CAPSULE V

REACTION AND AXIAL LOCATION	RADIAL LOCATION (cm)	SATURATED ACTIVITY (DPS/mg)	ADJUSTED SATURATED ACTIVITY (DPS/mg)	FAST NEUTRON FLUX (n/cm^2-sec) CAPSULE V	CALCULATED
$Fe^{54}(n,p)Mn^{54}$					
Top	192.46	2.45×10^3	2.89×10^3	5.35×10^{10}	
Top middle	192.46	2.84×10^3	3.35×10^3	6.20×10^{10}	
Middle	192.46	2.68×10^3	2.69×10^3	5.85×10^{10}	
Bottom middle	192.46	2.63×10^3	3.10×10^3	5.74×10^{10}	
Bottom	192.46	2.95×10^3	3.49×10^3	6.45×10^{10}	
Average				5.92×10^{10}	4.84×10^{10}

REFERENCES: SECTION 5.3

5.3.1-1 Norris, E. B. "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit 2, Analysis of Capsule V," Southwest Research Institute - Final Report SWRI Project No. 02-4397.

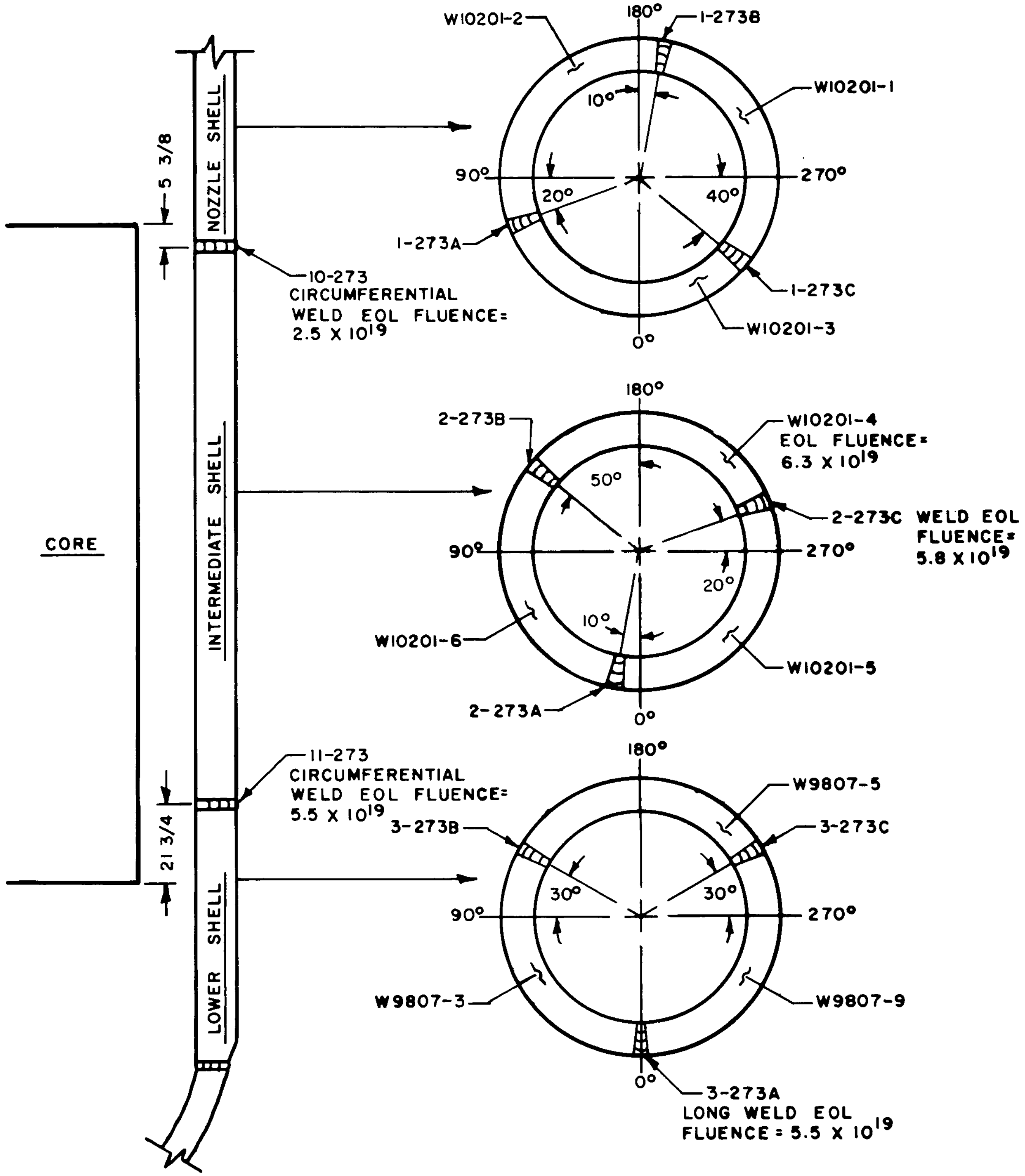
5.3.1-2 Meyer, T. A., "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants," WCAP - 10019, December, 1981.

5.3.1-3 C. E. Power Systems (L. C. Miller) letter to Carolina Power & Light Company (B. Furr), "Backwelding of Longitudinal and Circumferential Weld Seams and Transmittal of Typical Weld Metal Chemical Analysis," December, 1981.

5.3.1-4 Carolina Power & Light Company letter, "Thermal Shock to Reactor Pressure Vessels," E. E. Utley to D. G. Eisenhut (NRC), January 25, 1982, with attachments.

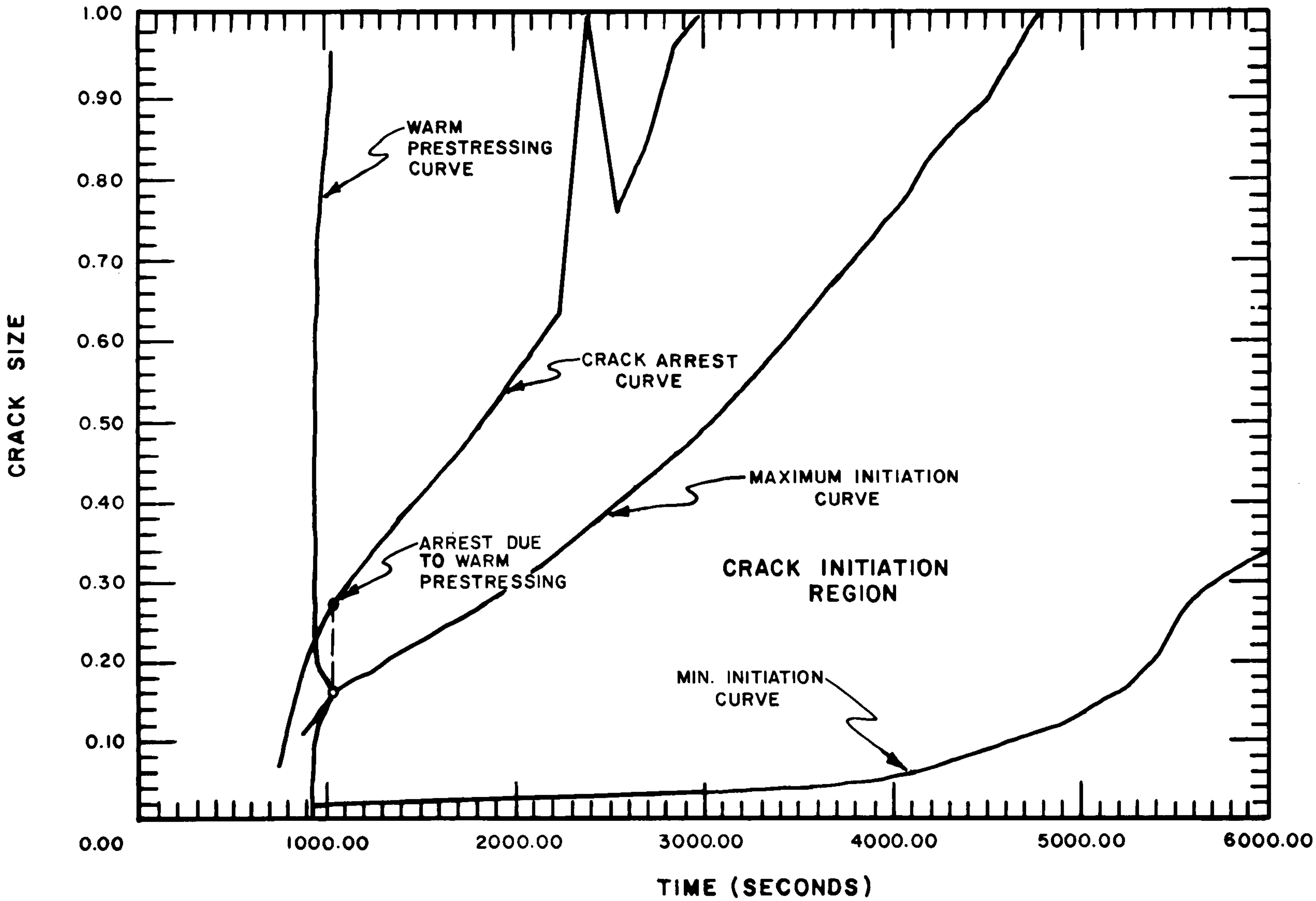
5.3.1-5 Yanichko, S. E., "Carolina Power & Light Company, H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy System - WCAP-7373 (January, 1970).

5.3.1-6 SRI letter to Carolina Power & Light Co. (Revision to Reference 5.3.1-1), E. B. Norris to T. Clements, April 4, 1977.



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE REACTOR VESSEL	FIGURE 5.3.1 - 1

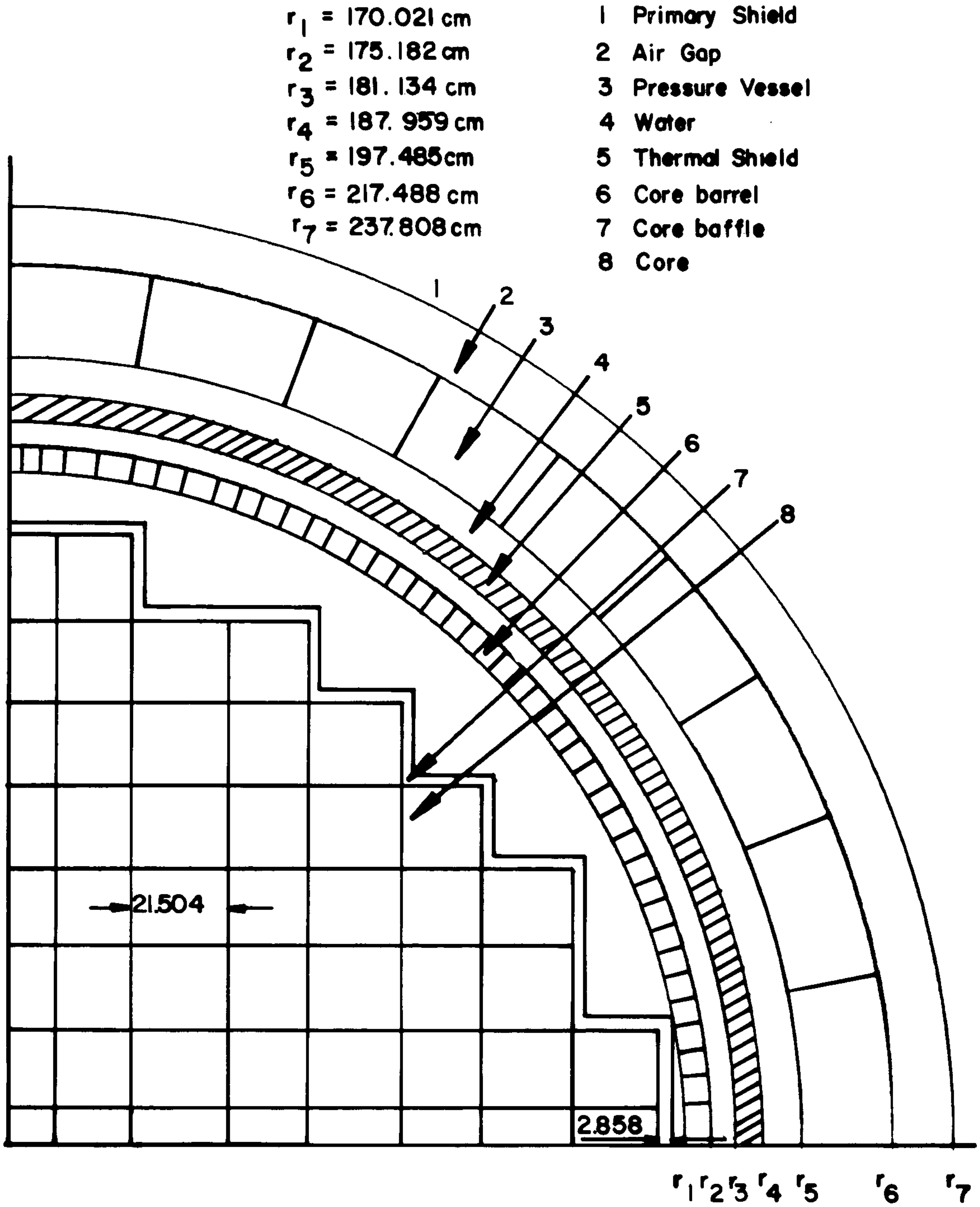
H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL
SAFETY ANALYSIS REPORT



CRITICAL CRACK SIZES

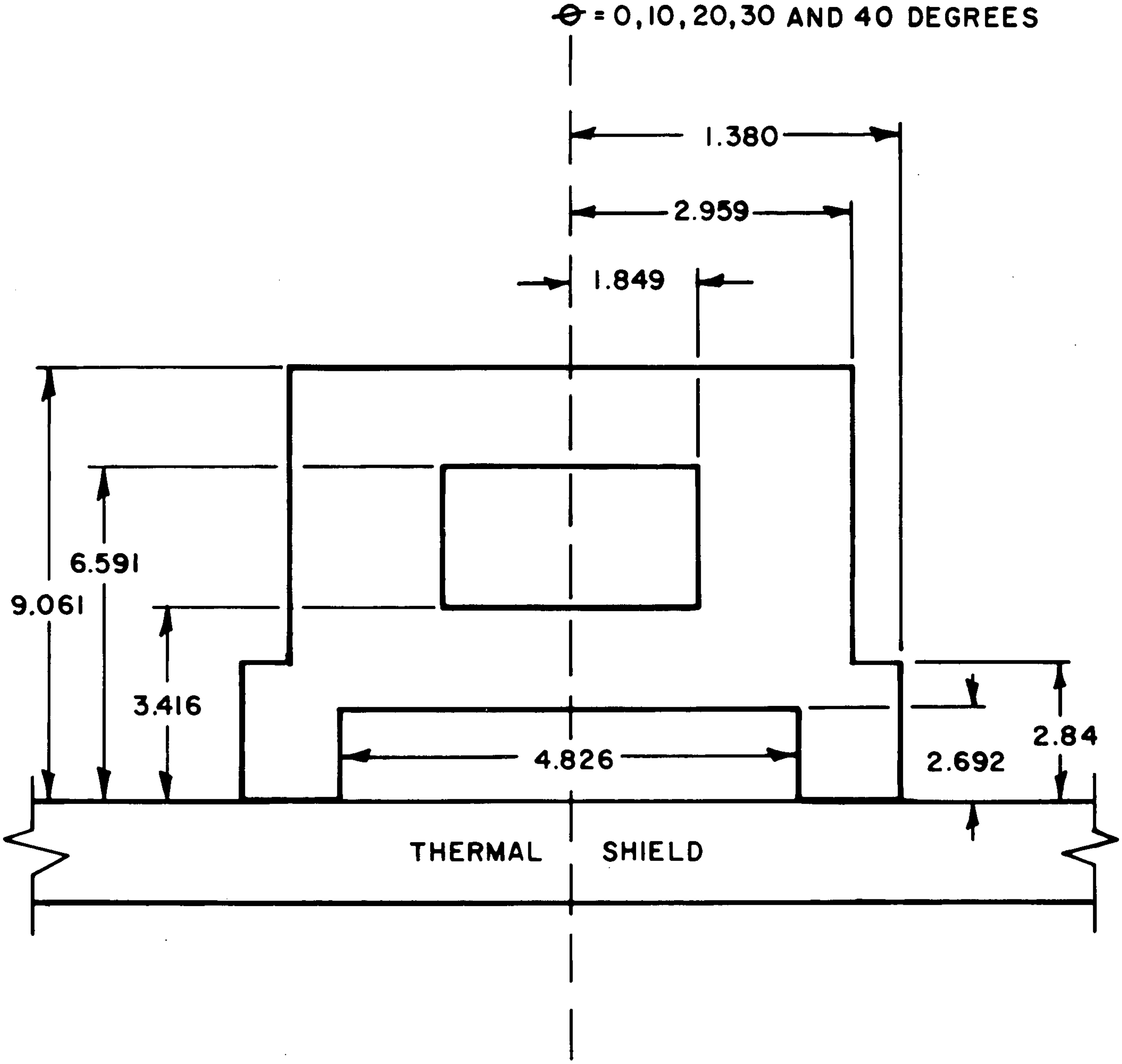
EFFECT OF WARM PRESTRESSING

FIGURE 5.3.1-2



(r, θ) Reactor Geometry

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PLAN VIEW OF REACTOR AT CORE MIDPLANE	FIGURE 5.3.1 - 3



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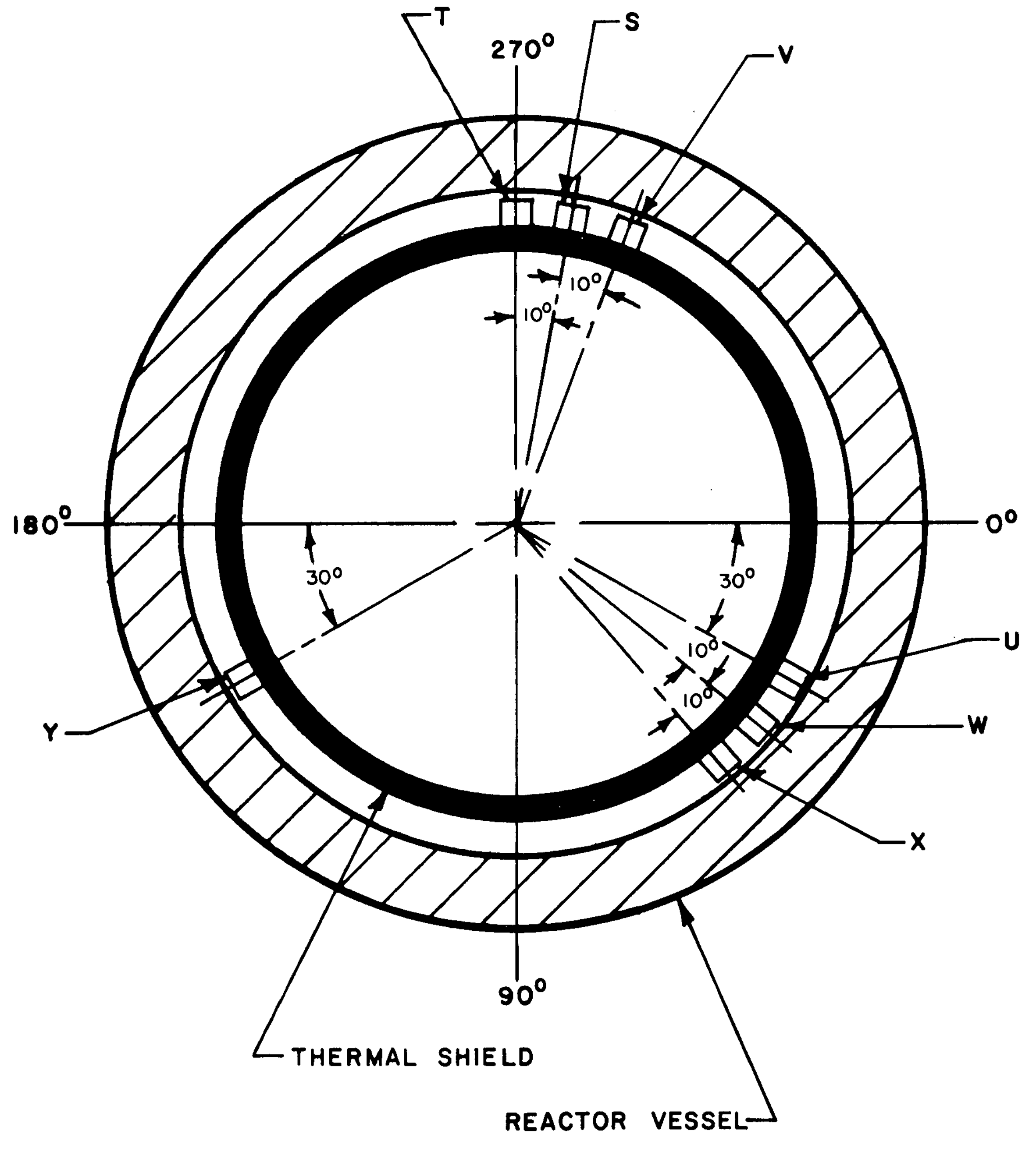
SURVEILLANCE CAPSULE DIMENSIONS (cm)

FIGURE
5.3.1 - 4



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	ARRANGEMENT OF SURVEILLANCE CAPSULES	FIGURE 5.3.1 - 5

TABLE 5.4.2-1

STEAM GENERATOR DESIGN DATA (2300 MW)

Number of Steam Generators	3
Design Pressure, Reactor Coolant/Steam, psig	2485/1085
Reactor Coolant Hydrostatic Test pressure	
(tube side-cold), psig	3110
Design Temperature, Reactor Coolant/Steam, ^{o}F	650/556
Reactor Coolant Flow, lb/hr	33.8×10^{6}
Total Heat Transfer Surface Area, ft^2	44,430
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, lb/hr	3.196×10^{6}
Steam Temperature, ^{o}F	516
Steam Pressure, psig	790
Feedwater Temperature, ^{o}F	420
Overall Height, ft-in.	63-1.6
Shell OD, upper/lower, in.	166/127.5
Shell Thickness, upper/lower, in.	3.5/2.63
Number of U-tubes	3260
U-tube Diameter, in.	0.875
Tube Wall Thickness, (average), in.	0.050
Number of Manways/ID, in.	3/16
Number of handholes/ID, in.	2/6

6.1.1.1.2 Containment Spray System Components

Containment Spray System components in contact with borated water, the sodium hydroxide spray additive, or mixtures of the two, are stainless steel or an equivalent corrosion-resistant material.

The principal components of the Containment Spray System consist of two pumps, one spray additive tank, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the spray additive tank are located in the Auxiliary Building and the spray pumps take suction directly from the refueling water storage tank. As all of the active components of the Containment Spray System are located outside the containment, they are not required to operate in the steam-air environment produced by a hypothetical accident.

The Containment Spray System also utilizes the two residual heat removal pumps, two residual heat exchangers and associated valves and piping of the SI System for the long-term recirculation phase of containment cooling and iodine removal (refer to Section 6.1.1.1.1).

The containment spray pumps were designed in accordance with the specifications discussed in Section 6.1.1.1.1 for the pumps in the SI System. The materials of construction are stainless steel or equivalent corrosion-resistant material.

The piping for the Containment Spray System was designed in accordance with the specifications discussed for the piping in the SI System (Section 6.1.1.1.4).

Spray nozzles and piping were built to conform to USAS B31.1. Nozzles are constructed of stainless steel.

The valves for the Containment Spray System were designed in accordance with the specifications discussed for the valves in the SI System, and conformed to the criteria of USAS B16.5. Valving descriptions and valve details are shown in Section 6.5.2.

The spray additive tank was constructed of austenitic stainless steel, and conformed to the requirements of the ASME code, Section III, Class C.

The Containment Spray System shares the refueling water storage tank liquid capacity with the SI System. Refer to Section 6.1.1.1.1.7 for a description of this tank.

6.1.1.1.3 Containment Air Recirculating System Components

All fan parts, damper shaft, and blade seating surfaces and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation and bearings were designed for operation during accident conditions.

The coils are fabricated of copper plate fins vertically oriented on copper tubes.

Ducts are constructed of corrosion-resistant material. Where flanged joints are used, joints are provided with gaskets suitable for temperatures to 300°F.

6.1.1.1.4 Post-Accident Containment Venting System Components

The Post-Accident Containment Venting System consists of two full capacity supply lines through which hydrogen-free air can be admitted to the containment; two full capacity exhaust lines through which hydrogen bearing gases may be vented from the containment; and, associated valving and instrumentation. The supply lines use equipment and piping which provide instrument air and service air during normal operation. One of the exhaust lines uses equipment and piping which normally provides pressure relief for the containment. The second exhaust line does not use existing equipment. Equipment added to permit the post-accident venting process was constructed of stainless steel. Details of the Post-Accident Containment Venting System are provided in Section 6.5.3.

6.1.1.1.5 Containment Structural Components

As discussed in Section 3.8.1.6, basically eight materials, of which six are metallic, have been used for construction of the containment structure. Metallic materials and components of the containment are as follows:

a) Reinforcing steel

b) Prestressed Steel System

c) Plate steel penetration frame

d) Liner

e) Equipment hatch and personnel lock, and

f) Pipe piles.

Metallic materials used for pipe piles are discussed in Section 3.8.5.

Metallic materials used for reinforcing steel, the prestressed steel system, the plate steel penetration frames, the liner, the equipment hatch, and personnel lock are discussed in Section 3.8.1.6.

6.1.1.1.6 Isolation Valve Seal Water System Components

The Isolation Valve Seal Water System provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed diaphram-type isolated valves (refer to Section 6.8 for the system design description).

The piping and valves for the system, including the air-operated valves, were designed in accordance with the USAS Code for Pressure Piping (Power Piping System), B31.1.

The isolation valve seal water tank was constructed of ASTM A-240, in accordance with the criteria of the ASME Code, Section VIII. The design data for the tank are given in Table 6.1.1-1.

There are no components of this system located inside containment.

c) Residual heat production

d) Metal-water reaction energy, and

e) Resulting hydrogen-oxygen reaction energy.

All the power generated by the core during blowdown is transferred to the coolant, and reaches the containment. The initial core stored and metal sensible energy is transferred to the coolant by a time dependent temperature difference calculation. It should be emphasized that the energy transferred from the core to the coolant for the containment evaluation far exceeded that transferred from the core thermal evaluation. That is to say, a conservatively high core heat transfer coefficient was used for the containment evaluation, while a conservatively low coefficient was used during the core thermal evaluation. Between the end of blowdown and the beginning of core reflooding there is no energy entering the containment. While the core is being reflooded the remaining stored energy in the core and internals causes a portion of the accumulator water to be boiled, and this energy is transferred to the containment.

Any energy addition resulting from a $Zr-H_2O$ reaction was also considered. The reaction energy reaches the containment by transfer to coolant, while the recombination energy of the H_2 generated in the reaction is added directly to the steam-air mixture in the containment. The hydrogen is assumed to burn as it is produced.

Finally, hot metal surfaces not cooled by SI water (reactor vessel above nozzles and SG tubes) were simulated as hot walls in contact with the containment steam-air mixture. A small heat transfer coefficient is employed to reflect actual conditions since these surfaces are covered by stagnant steam inside the RCS.

The following are some additional conservative assumptions used in the analysis:

a) The reactor power is based on operation at the maximum calculated power of 2300 Mwt

b) The decay heat is based on power operation for an infinite time

c) Coolant temperatures are the maximum levels attained in steady state operation, including allowance for instrument error and deadband

d) Gross system volumes are calculated from component dimensions, to which is added a 3 percent margin, and

e) Pressurizer liquid inventory at the nominal full power level plus an appropriate margin for instrument error and deadband.

The energy sources presented in Table 6.2.1-1 are potentially available to be transferred to the containment during the blowdown time.

The integrated energy balance at the end of blowdown is presented in the Table 6.2.1-3. The values were determined by the FLASH R Code.

In the above energy summation all sensible energy sources are referenced to the datum of saturated water at containment design pressure, which is the maximum amount of energy that can be transferred from the metal to the coolant.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures

The design pressure and temperature of the HBR 2 containment structure were those created by the hypothetical LOCA.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of ECCS

Section 6.2.1.1.3, Design Evaluation, presents the results of perturbations in mass and energy release to determine the effectiveness of ECCS for HBR 2.

6.2.1.6 Testing and Inspeccion

Tests performed on materials and special construction techniques are described in Section 3.8.1.6. Structural integrity tests of the completed Containment Building are described in Section 3.8.1.7. The in-service inspection program for associated ESF components is discussed in Section 3.9.

6.2.1.7 Instrumentation

Instrumentation has been provided to monitor containment atmospheric conditions:

Pressure	-5 to 126 psig
Radiation	Up to 10^7 R/hr photon
Hydrogen Concentration	0 to 10 percent
Water Level	Up to 600,000 gallons

Containment pressure indication will be used to distinguish between various incidents. Pressure taps reflect the effectiveness of the containment and cooling systems and other ESF. High pressure indicates high temperatures and reduced pressure indicates reduced temperatures. Indicators and alarms are provided in the Control Room to inform the operator of system status and to guide actions taken during recovery operations.

Detailed descriptions for all containment instrumentation, including diversity and redundancy considerations, are provided in Section 7.3.

The ESF systems are actuated by the ESF actuation channels. Each coincidence network energizes an ESF actuation device that operates the associated ESF equipment, motor starters, and valve operators. The channels are designed to combine redundant sensors, and independent channel circuitry, coincident trip logic, and different parameter measurements so that a safe and reliable system is provided in which a single failure will not defeat the channel function. The action initiating sensors, bistables, and logic are shown in the figures included in the detailed ESF Instrumentation Description given in Section 7.3.

The ESF actuation circuits are designed on the same "de-energize to operate" principle as the reactor trip circuits with the exception of the containment spray actuation circuit which is energized to operate in order to avoid spray operation on inadvertent power failure.

The spray system will be actuated by the coincidence of two sets of two out of three (Hi-Hi) containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header. The valves associated with the spray additive tank will be opened automatically.

The operator can manually actuate the entire system from the Control Room, and periodically, the operator will actuate system components to demonstrate operability.

The containment air recirculation coolers are normally in use during plant operation. These units are in the automatic sequence which actuates the ESF upon receiving the necessary signals indicating an accident condition, e.g., a high containment pressure signal automatically actuates the SI safety feature sequence which trips any closed inlet butterfly valves to the open position, trips any open inlet dampers to the closed position and starts any stopped fan cooler unit.

ESF Instrumentation Equipment - The following instrumentation ensures monitoring of the effective operation of the ESF.

Containment Pressure - Six channels, monitoring containment pressure, and derived from three pressure taps, reflect the effectiveness of the containment and cooling systems and other ESF. High pressure indicates high temperatures and reduced pressure indicates reduced temperatures. Indicators and alarms are provided in the Control Room to inform the operator of system status and to guide actions taken during recovery operations. Containment pressure indication will be used to distinguish between various incidents.

Redundant containment pressure signals are provided to isolate the containment. The containment pressure is sensed by six separate pressure transmitters located outside the containment. Containment pressure is communicated to the transmitters through three 1 in. stainless steel lines penetrating the containment vessel.

Each of the three pairs of differential pressure transmitters external to the containment in the Auxiliary Building have their own connection to the containment. Remote indicating facilities, and alarm and control signals are provided from each transmitter.

Remote indicating facilities have been provided which afford the operator the opportunity to read containment pressure.

Refueling Water Storage Tank Level - Level instrumentation on the refueling water storage tank consists of two channels. One channel provides a local indication and low level alarm function. The second channel provides remote indication (on the control board) and two low level alarms. One of these is a normal operating low level and the other is a low-low level alarm.

Containment Spray Flow - Instrumentation monitoring containment spray and additive flow is described in Section 6.5.2.5.

Pump Energization - All pump motor power feed breakers indicate that they have closed by energizing indicating lights on the control board.

Valve Position - All ESF remote-operated valves have position indication on the control board to show proper positioning of the valves. Air-operated and solenoid-operated valves are selected to move in a preferred direction with the loss of air or power. After a loss of power to the motors, motor-operated valves remain in the same position as they were prior to the loss of power.

Air Coolers - The cooling water discharge flow and exit temperature of each of the coolers are alarmed in the Control Room if the flow is low or if the temperature is high. The transmitters are outside the reactor containment. In addition, the exit flow is monitored for radiation and alarmed in the Control Room if high radiation should occur. This is a common monitor and the faulty cooler can be located locally by manually valving each one out in turn.

Sump Instrumentation - The containment sump instrumentation consists of two five-point level switches with gasketed junction boxes designed to operate in a post-accident environment. The transmitter housings are located above any possible flooding level. The indicators and alarm system are located in the Control Room.

Alarms - Visual and audible alarms are provided to call attention to abnormal conditions. The alarms are of the individual acknowledgement type; that is, the operator must recognize and silence the audible alarm for each alarm point.

b) Operating temperature can exceed 212°F, and

c) The fluid is radioactive.

All air and motor operated containment isolation valves can be remotely operated from the central Control Room. The open or closed conditions of these valves are displayed visually in the Control Room at all times.

Only the valves located inside the containment which were missile protected can be considered as available for containment isolation. These valves were located outside the missile barrier.

All lines penetrating the containment which normally carry radioactive fluids or that can communicate with the containment atmosphere following an accident were provided with radiation shielding in all areas where personnel access is possible. Manual valves in the lines, including containment isolation valves, were equipped with extension handles for operation from outside the shielding. Manually operated valves in the non-radioactive seal water injection lines were located outside the shielding.

Valves that are normally open during power operation and which must be closed for containment isolation are actuated to the closed position on receipt of a containment isolation signal.

Redundant electrical control circuits were provided for all remotely operated containment isolation valves. If the normal power supply for the control circuits fails, they may be energized by an emergency power supply. Duplicate cabling to the valve operators was not provided.

All air operated isolation valves fail closed on loss of control signal or control air. This is not detrimental to power operation. If one of the isolation valves should fail closed, operation of the connected systems either is not affected or can be modified until repairs are made.

It was necessary to demonstrate that containment isolation barriers were leak-tight. The closed systems that back up the containment isolation valves have adequate capability for flow toward the containment or adequate design to contain any radioactivity introduced into the system as the result of an accident. The water seal maintained between certain closed isolation valves by seal water injection was designed to prevent leakage of containment atmosphere to the environment by ensuring that any leakage through the valve seats or past stem packing is seal water, not containment atmosphere.

In general, vertical water legs were not used to seal the closed isolation valves. However, on lines isolated by two remotely operated valves in series, a loop seal or vertical water leg was installed between the isolation valves and the containment. This prevents loss of the water seal provided by seal water injection if the first outside isolation valve fails to close and the line is exposed to the containment atmosphere. Presence of water in the loop seal or vertical leg is assured by the inflow of seal water.

Penetrating lines other than those associated with the engineered safety features (ESF) which continue to be used, at least for a time, after containment isolation include:

a) Main steam headers

b) Auxiliary feedwater headers

c) Reactor coolant pump cooling water supply lines

d) Reactor coolant pump cooling water return lines

e) Reactor coolant pump seal water supply lines

f) Containment air sample in if containment pressure <5 psig, and

g) Containment air sample out if containment pressure <5 psig.

Automatic isolation valve sizes are listed in Table 6.2.4-2.

6.2.4.3 Tests and Inspections

The HBR 2 containment structure was designed such that the maximum allowable containment vessel leakage rate shall not exceed 0.1 percent per day of the containment atmosphere at 42 psig and 263°F which are the maximum conditions of the DBA.

Leakage from the containment to the outside could occur in the following locations:

a) Containment Penetrations (L_{pen})

b) Containment Liner Welds (L_c)

c) Containment Liner Plates (L_L), and

d) Containment Isolation Valves (L_{iso}).

The leakage from the penetrations (L_{pen}) is continuously monitored by the PPS Section 6.9. The PPS also pressurizes several volumes formed by double containment isolation valves (see Table I) or by double gasketed seals. These include the spaces between butterfly type isolation valves in the purge supply and exhaust lines, containment pressure and vacuum relief lines, the double isolation valves in the containment radiation monitor inlet and outlet lines, and into the spaces formed by double gaskets in the fuel transfer tube and on the equipment hatch and personnel lock doors. Leakage designated by L_{pen} was defined to include leakage from these volumes as well as from the penetration sleeves. In this context the word "penetration" also includes these volumes. The pressure in these volumes was maintained above accident pressure (42 psig) at all times, thus assuring that no leakage from the containment to the outside can occur through these paths. The PPS is used to perform a sensitive leak rate test of these volumes to verify that leakage to the outside does not exceed the design limits.

Operation will continue on the basis that venting will be performed daily as required to maintain the hydrogen concentration at approximately 3 percent by volume. Air will be added by the supply system as required.

6.2.5.3 Design Evaluation

Several venting schemes have been analyzed, and the offsite doses resulting from venting have been calculated. In the reference case, venting is begun when the hydrogen concentration reaches 3 percent by volume. The venting would be started regularly at 1:00 P.M. every day, regardless of the weather conditions, and continued for about one hour. A venting system flow rate of 240 cfm would be required for this case. For the reference case assumptions, the hydrogen concentration would reach 3 percent in 54 days. In summary, if venting is carried out in the manner used in the reference case, that is, without the selection of weather conditions and with other conservative assumptions, the expected thyroid doses are 680 mrem at the site boundary and 1.9 mrem at the low population zone. The whole body doses for the reference case are 81 mrem at the site boundary and 0.23 mrem at the low population zone. The doses actually expected from the venting are much lower than those given above, since many factors of conservatism have been combined (Reference 6.5.2-1).

Details of the venting system, the analysis of hydrogen production and accumulation, meterological data, vent schedules and offsite doses have been submitted to the Nuclear Regulatory Commission (NRC) (Reference 6.2.5-1).

6.2.5.4 Tests and Inspections

Inservice inspection requirements for the Post-Accident Containment Venting System are discussed in Section 3.9.

6.2.5.5 Instrumentation Requirements

6.2.5.5.1 Hydrogen Concentration

A containment hydrogen monitoring system is capable of measuring the hydrogen concentration in the containment atmosphere continuously over the range 0 to 10 percent hydrogen when the containment is within -4.7 psig to 42 psig.

6.2.5.5.2 Temperature Indicators

Temperature indicators located in the exhaust lines give local readout of the temperature of the containment air being exhausted.

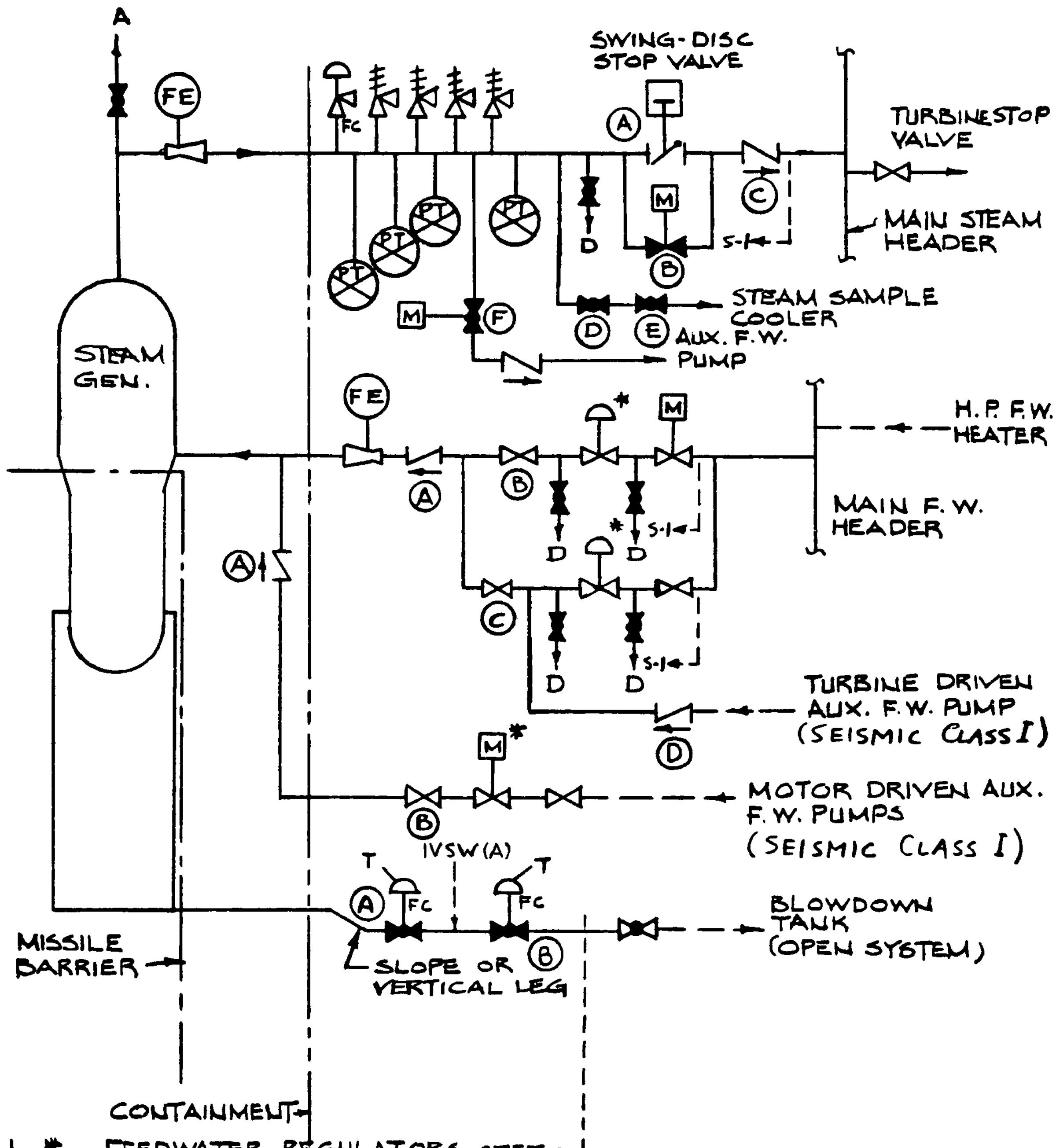
6.2.5.5.3 Flow Indicators

Flowmeters located in the supply and exhaust lines provide remote readout of process flow rates. The exhaust line instruments also indicate the total integrated flow vented from the containment.

6.2.5.5.4 Pressure Indicators

Local pressure indicators located in the exhaust lines are used to determine that the containment pressure is suitable for operating the venting system.

PENE. NOS. 7, 8, 9 – MAIN STEAM HEADERS
PENE. NOS. 10, 11, 12 – FEEDWATER HEADERS
PENE. NOS. 13, 14, 15 – STEAM GENERATOR BLOWDOWN LINES
PENE. NOS. 57, 58, 59 – EMERGENCY FEEDWATER HEADERS



1. * – FEEDWATER REGULATORS STET

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES STEAM AND FEEDWATER HEADERS	FIGURE 6.2.4 - 3

6.3.2 SYSTEM DESIGN

6.3.2.1 Schematic Piping and Instrument Diagrams

The SIS flow diagrams are shown in Figures 6.3.2-1 and 6.3.2-2. The initiating systems for SIS are discussed in Section 7.3.

6.3.2.2 Equipment and Component Descriptions

6.3.2.2.1 Injection Phase

The SI signal opens the SIS isolation valves and starts the SI pumps and the residual heat removal (RHR) pumps. The items on Figures 6.3.2-1 and 6.3.2-2 marked with an "S" receive the SI signal. The high head SI pumps take suction from the refueling water storage tank (RWST) and, since the boron injection tank (BIT) is located at the discharge of the high head SI pumps, its contents with 12 percent boric acid concentration will rapidly enter the RCS.

The RHR pumps deliver to all three cold legs through the piping between the accumulators and the cold legs. The high head SI pumps deliver automatically into a header connected to the cold legs and one connected to the hot legs. The header to the cold legs contains the BIT. Downstream of the BIT, the header divides into three injection lines connecting to the pipes from the accumulators close to the RCS cold leg piping. The capability is provided to manually isolate the pumps on separate headers from the reactor turbine generator board (RTGB), thereby ensuring the delivery of full flow from at least one pump for the special case of a broken header.

For large pipe breaks, the RCS would be depressurized and voided of coolant rapidly (about 10 sec for the largest break). A high flow rate is required to quickly recover the exposed fuel rods and limit possible core damage. To achieve this objective, one RHR pump (high flow, low head) is required to deliver borated water to the cold legs of the reactor coolant loops. Two pumps are available in order to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge.

The BIT, which is located in the high head SI header that discharges into the cold legs, initially provides injection of water at a high boron concentration. The tank contains boric acid at a nominal concentration of 20,000 ppm boron (12 percent boric acid solution) and is isolated from the RCS by two check valves in series with parallel disk, normally closed, motor-operated gate valves. It is isolated from the accumulators by one check valve in series with the normally closed gate valves. The tank is isolated from the high head SI pumps by two parallel discs, normally-closed, motor-operated gate valves. Leakage of liquid at different concentration or chemistry into the BIT or leakage out of the tank is prevented by an inter-connecting line between the tank isolation valves. With this arrangement, the pressure differential across the tank when the valves are closed is zero.

Because the injection phase of the accident is terminated before the RWST is completely emptied, all pipes are kept filled with water before recirculation is initiated. Water level indication and alarms on the RWST give the operator ample warning to terminate the injection phase. Additional level indicators and alarms are provided in the containment sump, which also give backup indication when injection can be terminated and recirculation initiated.

For small pipe breaks, the depressurization of the RCS by the SIS can be augmented by dumping steam to the atmosphere or the condensor, and addition of auxiliary feedwater to the SG. Use of steam dump is not required to meet the core cooling objectives. It is intended that for small breaks (4 in. and smaller), steam dump will be employed to facilitate the recovery from the accident, and to reduce the reactor coolant system pressure to the cut-in pressure of the RHR pumps.

The main steam isolation valves do not get a containment isolation signal. However, they do close automatically on a steam line break.

Since leakage between the RCS and the SG during operation is possible, careful consideration is given to the effect of any possible radioactive leakage into the SG. Manual steam dump to the atmosphere will not be initiated unless it can be assured that there has been no measured contamination of the SG as a result of the LOCA.

Breaks large enough to release fission products from the core are characterized by a rapid depressurization of the RCS and uncovering of the core, followed by an increase in fuel clad temperature causing the cladding to burst. For these breaks, the reactor coolant pressure would fall below that of the SG before the SG is pressurized to the SG safety valves' setpoint. There would be no leakage of radioactivity to the atmosphere.

Before initiating any cooldown of the SG either by atmospheric steam dump or steam dump to the condenser, the operator would check the activity in the SG by use of the radiation monitor in the SG blowdown line. The auxiliary operator would open the blowdown lines one at a time from the local control station and the Control Room operator would observe the readings on the radiation monitor. If the readings showed an increase over the normal operating level, steam dump would not be permitted and the SG would remain isolated for the duration of the accident. Plant laboratory technicians would also sample each SG for activity.

When steam dump cooldown is used for small breaks (4 in. and smaller), the steam will be dumped to the condenser when outside power is available, or directly to the atmosphere when outside power is not available. The expected peak fuel clad temperatures for break sizes 4 in. and smaller are limited to a value below which cladding bursting is expected. When steam dump is initiated, the only activity that could be leaked into the steam would be dumped to the condenser if outside power is available. In that case, the air ejector radiation monitor would provide additional information that activity carryover to the secondary side had not occurred as a result of the accident.

6.3.2.2.2 Recirculation Phase

After the injection phase of SIS operation, coolant spilled from the break and water collected from the containment spray is cooled and returned to the RCS by the recirculation system.

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INSTRUMENTATION AND CONTROLS

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average signal is conditioned to provide an analog voltage signal for use in permissive control and protection bistable amplifiers.

Isolation amplifiers, which provide remote control signals and core power status information to the operator and computer, also utilize the average power analog signal. The four power range channels are operated from separate AC sources and are housed in separate racks so that a single failure will not cause loss of protection functions. Redundant relays for the protection functions are located in the logic portion of the protection system.

Isolated analog outputs from the power range channels are compared in a separate auxiliary channel drawer. This comparator provides the operator with annunciation of deviations in average power between the four power range channels. Switches are provided to defeat this comparison for a failed channel so that subsequent deviations or failures among the three remaining channels are annunciated.

7.2.1.1.7.4 Detectors

The NIS employs six detector radial locations containing a total of eight detectors (two proportional counters, two compensated ionization chambers and four dual section uncompensated ionization chamber assemblies) installed around the reactor in the primary shield. Windows in the primary shield minimize leakage flux attenuation and distortion.

Boron-10 lined proportional counters having a nominal thermal neutron sensitivity of ten counts per neutron per square centimeter per second, provide pulse signals to the source range channels. These detectors are installed on opposite "flat" portions of the core containing the primary startup sources, at an elevation approximating the quarter core height.

Compensated ionization chambers serve as neutron sensors for the intermediate range channels and are located in the same instrument wells and detector assemblies as the source range detectors. These detectors have a nominal thermal neutron sensitivity of 4×10^{-14} amperes per neutron per square centimeter per second. Gamma sensitivity is less than 3×10^{-11} amperes per Roentgen per hour when operated uncompensated, and is reduced to approximately 3×10^{-13} amperes/R/hr in compensated operation. The detectors are positioned at an elevation corresponding to the center of the quarter core height.

The detector assemblies containing one each of the above-mentioned detectors use watertight, corrosion-resistant steel enclosures. High density polyethylene, used as a moderator-insulator within the detector assemblies, will be confined at temperatures associated with a loss-of-coolant accident (LOCA). The detectors are connected to the junction box at the top of the detector well by special high temperature, radiation-resistant cables.

The remaining four detector assemblies contain the power range ionization chambers. Each provides two current signals corresponding to the neutron flux in the upper and lower sections of a core quadrant. These detectors have a total neutron-sensitive length of ten feet and a nominal thermal neutron sensitivity for each section of 1.7×10^{-13} amperes per neutron per square centimeter per second. Gamma sensitivity of each section is approximately 10^{-10} amperes per Roentgen per hour.

The detector assemblies for power range operation are installed vertically and located equidistant from the reactor vessel at all points, and, to minimize neutron flux pattern distortions, within one foot of the reactor vessel. Cabling from individual detector wells to the containment penetrations and to the instrument racks in the Control Room are routed in individual conduits, with physical separation between the penetrations and conduits associated with redundant protective channels.

7.2.1.1.7.5 Detailed Description

The source range output information is tabulated in Table 7.2.1-4. The detector for each source range channel is a Boron-10 lined proportional counter. The signal received from the counter has a range of 1 to 10^6 pulses per second randomly generated and is received through a fixed gain pulse preamplifier located outside the containment. The preamplifier optimizes the signal-to-noise ratio and also furnishes high voltage coupling to the detector.

The preamp has internal provisions for generating self-test frequencies of 60 counts per second (cps) and 10^6 cps. These test oscillator circuits are energized by a switch located on the associated source range drawer. The source range channel power supplies furnish low voltage for preamp operation as well as low voltage for the drawer-mounted modules. The preamp is solid state in design with discrete components and includes an impedance matching network between the preamp output and the 75-ohm triaxial cable.

The preamp output is received at the post-amplifier located on the source range drawer. This module provides amplification and discrimination, both of which are adjustable. Discrimination is provided between neutron flux pulses and combined noise and gamma-generated pulses. The discriminator supplies two outputs: one output (isolated) to a scaler-timer unit on the visual-audio channel drawer (see source range auxiliary equipment); and the other to a pulse shaper (transistorized flip-flop circuit) which supplies a constant amplitude pulse to the log integrator module within the source range drawer.

Logarithmic integration of the pulse signal is performed in another modular unit to obtain an analog DC signal. The log signal is then amplified for local indication on the front panel of the source range drawer, and is also delivered through a parallel run to the source range level bistables and isolation amplifier. The analog output signal is proportional to the count rate being received from the sensor and is displayed by the front panel meter on a scale calibrated logarithmically from 10^0 to 10^6 cps. The solid state isolation amplifier provides five analog outputs, all of which are adjustable through attenuator controls. Three outputs are used as follows: as remote indication (0-1 ma); as remote recording (0-37.5 mv DC); and as an input to the computer (0-5 V DC). A 0-10 V DC output is used by the startup-rate amplifier to produce a startup-rate indication at the main control board. The remaining output (0-5 V DC) is a spare.

7.2.2.2.3 Pressurizer Pressure

Three pressure channels are used for high and low pressure protection and as part of overpower-overtemperature protection. Isolated output signals from these channels also are used for pressure control and compensation signals for rod control. These are discussed separately below:

a) Control of rod motion: one of the pressure channels is used for rod control with a low pressure signal acting to withdraw rods. The discussion for coolant temperature is applicable, i.e., two-out-of-three logic for overpower-overtemperature protection as the primary protection, with backup from multiple rod stops and "backup" trip circuits. In addition, the pressure compensation signal is limited in the control system such that failure of the pressure signal cannot cause more than about $10^{o}F$ change in T_{avg}. This change can be accommodated at full power without a DNBR less than 1.30. Finally, the pressurizer safety valves are adequately sized to prevent system overpressure.

b) Pressure Control: Spray, power-operated relief valves, and heaters, are controlled by isolated output signals from the pressure protection channels.

1) Low Pressure - A spurious high pressure signal from one channel can cause low pressure by spurious actuation of spray and/or a relief valve. Additional redundancy is provided in the protection system to ensure underpressure protection, i.e., two-out-of-three low pressure reactor trip logic and two-out-of-three logic for safety injection.

2) High Pressure - The pressurizer heaters are incapable of overpressurizing the reactor coolant system. Maximum steam generation rate with heaters is about 11,000 lb/hr, compared with a total capacity of 864,000 lb/hr for the three safety valves and a total capacity of 420,000 lb/hr for the two power-operated relief valves. Therefore, overpressure protection is not required for a pressure control failure. Two-out-of-three high pressure trip logic is therefore used.

In addition, either of the two relief valves can easily maintain pressure below the high pressure trip point. The two relief valves are controlled by independent pressure channels, one of which is independent of the pressure channel used for heater control. Finally, the rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available for operator action.

7.2.2.2.4 Pressurizer Level

Three pressurizer level channels are used for reactor trip. Isolated output signals from these channels are used for volume control, increasing or decreasing water level. A level control failure could fill or empty the pressurizer at a slow rate (on the order of half an hour or more).

A reactor trip on pressurizer high level is provided to prevent rapid thermal expansions of reactor coolant fluid from filling the pressurizer: the rapid change from high rates of steam relief to water relief can be damaging to the safety valves and the relief piping and pressure relief tank. However, a level control failure cannot actuate the safety valves because the high

pressure reactor trip is set below the safety valve set pressure. With the slow rate of charging available, overshoot in pressure before the trip is effective is much less than the difference between reactor trip and safety value set pressures. Therefore, a control failure does not require protection system action.

In addition, ample time and alarms are available for operator action.

7.2.2.2.5 Steam Generator Water Level; Feedwater Flow

Before describing control and protection interaction for these channels, it is beneficial to review the protection system basis for this instrumentation. (See Figure 7.2.1-16.)

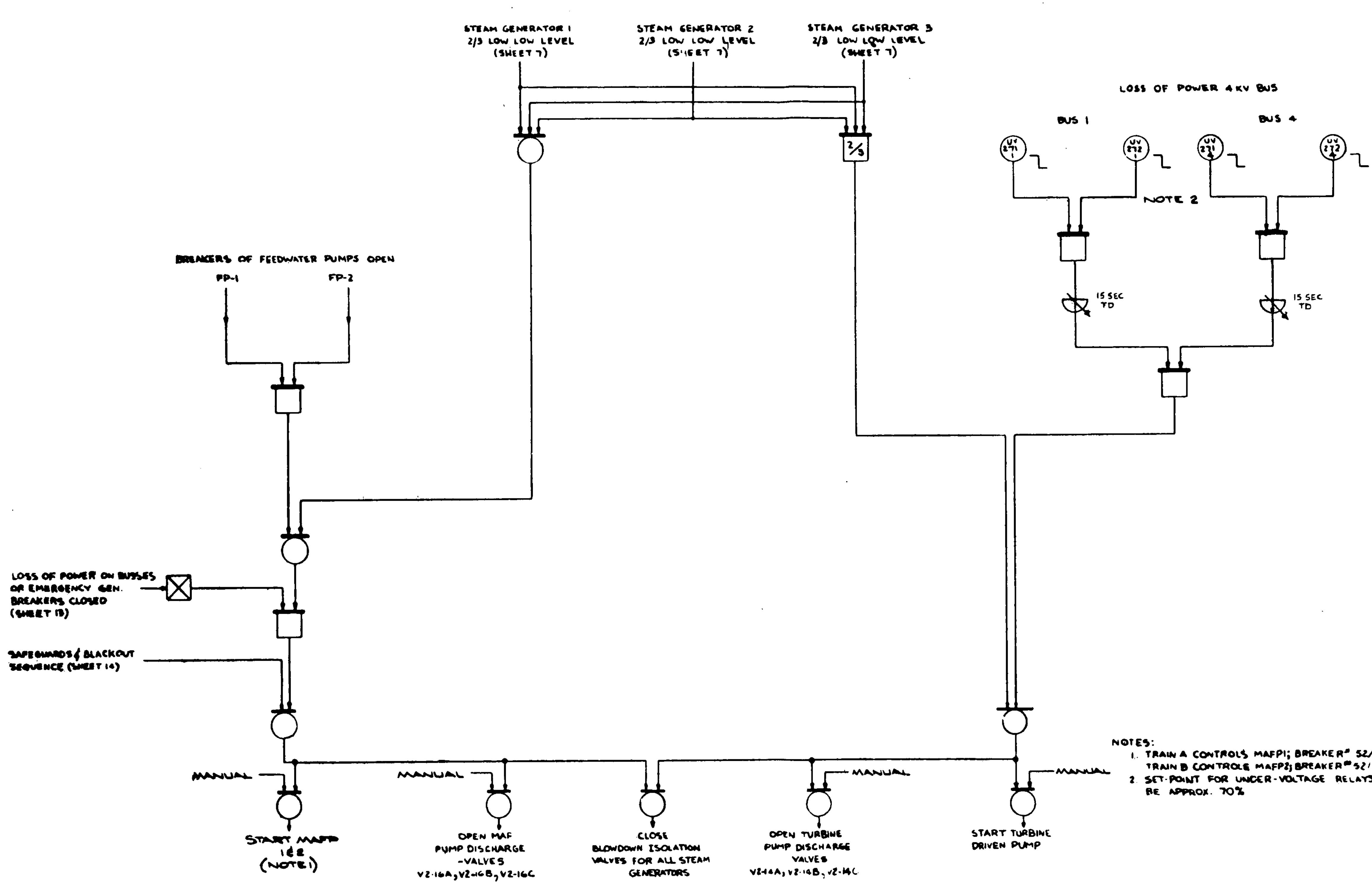
The basic function of the reactor protection circuits associated with low steam generator water level and low feedwater flow is to preserve the steam generator heat sink for removal of long-term residual heat. Should a complete loss of feedwater occur with no protective action, the steam generators would boil dry and cause an overtemperature-overpressure excursion in the reactor coolant. Reactor trips on temperature, pressure, and pressurizer water level will trip the plant before there is any damage to the core or reactor coolant system. However, residual heat after trip would cause thermal expansion and discharge of the reactor coolant to containment through the pressurizer relief valves. Redundant emergency feedwater pumps are provided to prevent this. Reactor trips act before the steam generators are dry to reduce the required capacity and starting time requirements of these pumps and to minimize the thermal transient on the reactor coolant system and steam generators. Independent trip circuits are provided for each steam generator for the following reasons:

a) Should severe mechanical damage occur to the feedwater line to one steam generator, it is difficult to ensure the functional integrity of level and flow instrumentation for that unit. For instance, a major pipe break between the feedwater flow element and the steam generator would cause high flow through the flow element. The rapid depressurization of the steam generator would drastically affect the relation between downcomer water level and steam generator water inventory.

b) It is desirable to minimize the thermal transient on a steam generator for credible loss of feedwater accidents.

It should be noted that controller malfunctions caused by a protection system failure affect only one steam generator. Also, they do not impair the capability of the main feedwater system under either manual control or automatic control. Hence, these failures are far from being the worst cases with respect to decay heat removal with the steam generators.

1) Feedwater Flow - A spurious high signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow and prevent that channel from tripping. A reactor trip on low-low water level, independent of indicated feedwater flow, will ensure a reactor trip if needed.



NOTES:

1. TRAIN A CONTROLS MAFP1; BREAKER # 52/20A
   TRAIN B CONTROLS MAFP2; BREAKER # 52/24C
2. SET-POINT FOR UNDER-VOLTAGE RELAYS SHOULD BE APPROX. 70%

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

LOGIC DIAGRAMS - SHEET 11
AUXILIARY FEEDWATER PUMP START-UP

FIGURE 7.2.1 - 27

7.3 ENGINEERED SAFETY FEATURE SYSTEMS

The Engineered Safety Feature (ESF) Instrumentation measures temperatures, pressures, flows, and levels in the Reactor Coolant System (RCS), Reactor Containment, and Auxiliary Systems, activates the ESF, Containment Isolation, Steam Line Isolation, and Emergency Feedwater, and monitors their operation.

7.3.1 DESCRIPTION

7.3.1.1 System Description

The ESF actuation instrumentation performs the functions shown in Table 7.3.1-1. These functions are summarized below.

a) Operation of the Safety Injection System (SIS) is initiated upon occurrence of any of the following events: low pressurizer pressure; high containment pressure; high differential pressure between any steam line and the steam line header; or high steam flow in any 2 steam lines, coincident with low steam pressure or low reactor coolant average temperature.

b) Operation of the containment isolation valves in nonessential process lines (phase A) is initiated upon automatic actuation of safety injection.

c) Operation of the Containment Spray System and remaining containment isolation valves (phase B) is initiated upon detection of a high-high containment pressure signal.

d) Operation of the Containment Air Recirculation Cooling System is started after initiation of the SIS.

e) The following signals will close all steam isolation valves:

1) High steam flow coincident with low reactor coolant average temperature or low steam pressure

2) High-high containment pressure signals

Steam line isolation is required to prevent the blowdown of more than one steam generator (SG) in the unlikely event of a steam line fracture.

f) Any safety injection signal will isolate the main feedwater lines by closing all control valves (main and bypass valves), tripping the main feedwater pumps and closing the pump discharge valves. The auxiliary feedwater system is actuated by the safety injection signal.

g) Although the emergency feedwater system is not considered to be an engineered safeguard, its actuation is described below.

7.3.1.1.1 Auxiliary Feedwater System Initiation

The controls used to automatically start the auxiliary feedwater pumps are designed to meet the single failure criterion. The following pump starting logic is used:

a) The two motor driven auxiliary feedwater pumps are started automatically on:

1) 2/3 low low level in any SG

2) Opening of both feedwater pump circuit breakers (one contact per pump breaker is used)

3) Any Safety Injection Signal

4) Loss of all AC power (i.e., the blackout sequence)

5) Manually

b) The turbine-driven auxiliary feedwater pump is started automatically on:

1) 2/3 low low level in any two SG

2) Loss of voltage on 4 kV buses 1 and 4. Two sensors are provided for each bus with 1/2 logic to indicate a loss of voltage on any one bus.

3) Manually

In the Loss of Normal Feedwater analysis, in Section 15, it has been assumed that the auxiliary feedwater pumps are started on the low low steam generator level signals. The analysis has been performed assuming only one motor-driven auxiliary feedwater pump is started at one minute after reaching the low low level setpoint in all three SG.

The relay logic for starting the auxiliary feedwater pumps is separated into train A and train B logic, as is done for the relay logic used to actuate ESF. Logic train A will start one motor driven pump and logic train B will start the second motor-driven pump. Either logic train will open appropriate steam system valves to start the turbine-driven pump. The circuits used to start the auxiliary feedwater pumps will also open the appropriate valves to ensure delivery of flow to the SG.

MALFUNCTION ANALYSIS

COMPONENT	MALFUNCTION	COMMENT
Motor Driven Pump	Fails to start	Second pump or turbine-driven pump supplies adequate feedwater
Steam Generator Level Switch	Fails to signal low Level	Three switches per generator provided. Two required for initiation.
Steam Admission Valve to Pump Turbine	Fails to open	Three valves, one from each SG
Discharge Valve From Motor-Driven Pump Header	Fails to open	If steam driven pump has not started, one steam generator will reach low level alarm point. Operator must manually start steam driven pump.

7.3.1.1.2 ESF Instrumentation

7.3.1.1.2.1 Design Bases

The ESF instrumentation measures temperatures, pressures, flows, and levels in the RCS, Steam System, Reactor Containment and Auxiliary Systems, actuates the ESF, and monitors their operation. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded, and controlled from the Control Room. The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems and processes over the full operating range of the plant.

Certain controls and indicators which require a minimum of operator attention, or are only in use intermittently, are located on local control panels near the equipment to be controlled. Monitoring of the alarms of such control systems are provided.

The same channel isolation and separation criteria as described for the reactor protection circuits (Section 7.2) are applied to the ESF actuation circuits.

The DC control supply associated with the ESF is designed to meet the single failure criterion such that one failure will not prevent actuation of sufficient ESF to meet the core and containment cooling criteria.

7.3.1.1.2.2 Design Features

The ESF instrumentation system is designed to use analog channels and initiation logic similar to that of the Reactor Protection System described in Section 7.2.

The initiating systems for the Containment Ventilation System, the Feedwater Isolation System, the Containment Spray System, and the Containment Isolation System are designed so that actuation cannot be over-ridden, and an open or failed reset switch will not impede manual operation. The Safety Injection Actuation System is designed so that an open or failed reset switch will not impede manual operation.

7.3.1.1.2.3 ESF Instrumentation Equipment

The following instrumentation ensures monitoring of the effective operation of the ESF.

a) Containment Pressure - Six channels, monitoring containment pressure, and derived from three pressure taps monitor the effectiveness of the containment cooling systems and other ESF. High pressure indicates high temperatures and reduced pressure indicates reduced temperatures. Indicators and alarms are provided in the Control Room to inform the operator of system status and to guide actions taken during recovery operations. Containment pressure indication will be used to distinguish between various incidents.

Redundant containment pressure signals are provided to isolate the containment. Each of the three pairs of differential pressure transmitters external to the containment in the Auxiliary Building have their own connection to the containment. Remote indicating facilities and alarm and control signals are provided from each transmitter.

Remote indicating facilities are provided which afford the operator the opportunity to read containment pressure.

b) Refueling Water Storage Tank Level - Level instrumentation on the refueling water storage tank consists of two channels. One channel provides a local indication and low level alarm function. The second channel provides remote indication (on the control board) and two low level alarms. One of these is a normal operating low level and the other is a low-low level alarm. The low level alarm has redundant alarm switches read from separate level transmitters and power supplies.

c) Emergency Core Cooling System Pumps Discharge Pressure - These channels clearly show that the emergency core cooling system pumps are operating. The transmitters are outside the containment.

d) Pump Energization - All pump motor power feed breakers indicate that they have closed by energizing indicating lights on the control board.

e) Radioactivity - Means are provided to measure the radioactivity in the containment atmosphere after the incident, since this information will be required for any subsequent entry into the containment following a loss-of-coolant accident (LOCA). In the event of a major LOCA, radioactivity levels would be such that area monitors located outside the containment would respond to the activity levels inside the containment. The containment system particulate and gaseous monitoring equipment would also provide information useful in post-accident recovery operations at pressures below 5 psig, and with favorable containment temperature and radiation conditions.

f) Valve Position - All ESF remote-operated valves have position indication on the control board to show proper positioning of the valves. Air-operated and solenoid-operated valves are selected to move in a preferred direction with the loss of air or power. After a loss of power to the motors, motor-operated valves remain in the same position as they were prior to the loss of power.

g) Air Coolers - The cooling water discharge flow of each of the coolers is alarmed in the Control Room if the flow is low. The transmitters are outside the reactor containment. In addition, the exit flow is monitored for radiation and alarmed in the Control Room if high radiation should occur. This is a common monitor and the faulty cooler can be located locally by manually valving each one out in turn.

h) Sump Instrumentation - The containment sump instrumentation consists of four level switches with gasketed junction boxes designed to operate in a post-accident environment. The transmitter housings are located above any possible flooding level. The indicators and alarm system are located in the Control Room.

Indicated on the reactor and turbine-generator board (RTGB) are two status lights which light when the water level in the reactor vessel cavity sump rises above 0.5 ft. The containment sump level is indicated on the RTGB from 0 to 7 ft above the containment floor in 0.5 ft increments. Two extended range (analog channels) level indicators are displayed on the core cooling and containment panel which indicate the water level from 3.5 in. above the reactor vessel cavity floor to 423.5 in.

i) Local Instrumentation - In addition to the above, the following local instrumentation is available.

1) Residual heat removal pumps discharge pressure

2) Residual heat exchanger exit temperatures

3) Containment spray test lines total flow

4) Safety injection test line pressure and flow

j) Alarms - Visual and audible alarms are provided to call attention to abnormal conditions. The alarms are of the individual acknowledgement type; that is, the operator must recognize and silence the audible alarm for each alarm point.

k) Indication - All transmitted signals (flow, pressure, temperature, etc.) which can cause actuation of the ESF features are either indicated or recorded for every channel.

7.3.1.1.2.4 Interlocks to Prevent Diesel Generator Overload During Safety Injection

a) Component Cooling Pumps - To limit the load on the diesel generators, the component cooling pumps are tripped upon coincident safety injection and blackout signal. However, the operator can maually restart the component cooling pump if a containment spray signal does not exist.

b) Pressurizer Heaters - The pressurizer heaters are fed from redundant diesel generator buses to assure natural circulation during loss of offsite power. However, upon initiation of a safety injection signal, the pressurizer heater load will be shed with the exception of 150 kW.

c) Safety Injection Block - During shutdown, the SG differential pressure safety injection signal is blocked during normal shutdown operation to prevent spurious safety injection due to large deviations in the SG pressure which normally occurs during plant shutdown.

7.3.1.2 Design Basis Information

The information presented in 7.2.1.2.2 is applicable. Additional design basis information is presented in 7.3.1.1, above.

7.3.1.3 Instrumentation Cable Separation

The Engineered Safety Features (ESF) System is divided into two channels with each channel run in individual cable tray systems throughout the plant. Cables from different channels are never routed through the same penetrations. These penetrations are grouped into two groups for channel 1 consisting of penetration C-3, D-2, and D-4 and channel 2 consisting of penetration B-8, D-8, and D-9. The penetration in these two groups are separated by a horizontal distance of approximatley 14 ft. Additional physical separation is provided by placing one complete channel consisting of penetration C-3, D-2, and D-4 on one side of a concrete wall separating this channel from channel 2 consisting of penetration B-8, D-8, and D-9.

The relays and associated circuitry for the ESF are located in the upper relay room in the southwest corner of the Reactor Auxiliary Building. The initiating systems are divided into four channels physically arranged so that each channel is at the extremity of two rows of process racks.

The rack arrangements and separation criteria are the same as that provided for the Reactor Trip System cable described in Section 7.2.2.3.

7.3.1.4 Final System Drawings

Figures 7.2.1-2, 7.2.1-3, 7.2.1-5, 7.2.1-6, 7.2.1-14, and 7.2.1-17 through 7.2.1-34 are applicable.

TABLE 7.3.1-1 (Cont'd)

AUXILIARY FEEDWATER ACTUATION	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
12. Turbine Driven Pump	Coincidence of 2/3 low level in two steam generators; or loss of voltage on 2/2 4kV volt buses; or manual 1/2	
13. Motor Driven Pumps	2/3 low level in any one SG; or trip of 2/2 main feedwater pumps; or safety injection signal; or manual 1/2	
MAIN FEEDWATER ISOLATION		
14. Close Main Feedwater Control Valves, Trip Main Feedwater Pumps	See Item 1.	

7.3.2 ANALYSIS

7.3.2.1 Instrumentation Used During Loss-Of-Coolant Accident

Instruments provided and designed to function following the major loss-of-coolant accident (LOCA) are those which initiate or otherwise govern the operation of ESF. Pressurizer pressure and SG flow and level sensors are located outside the missile barrier but inside the containment because an equivalent signal cannot be obtained from a sensor location more isolated from the reactor. SG pressure signals are located outside the containment.

It should be emphasized, however, that for the large LOCA, the initial suppression of the transient is independent of any detection or actuation signal. That is, the water level will be restored to the core by the passive accumulator system.

All pumps used for safety injection and initial containment spray are located outside the containment. The operation of the equipment can be verified by instrumentation that reads in the Control Room. This instrumentation will not be affected by the accident.

Depending upon the magnitude of the LOCA, information relative to the pressure of the RCS will be required to determine which pumps will be used for recirculation. The information relative to the pressure of the RCS will be required to decide if the charging pumps are required for make-up water, such as for a relatively small LOCA. Discharge pressure of the charging pumps, as read on instrumentation outside the containment, will be sufficient. In conjunction with the available accumulator instrumentation, a full range of system pressure can be determined.

The back-up for the instrumentation is the refueling water storage tank level instrumentation. Core recirculation and containment spray recirculation (if necessary) will be manually initiated, when the refueling water storage tank is empty.

Considerations have been given to all the instrumentation and information that will be necessary for the recovery time following a LOCA. Instrumentation external to the reactor containment such as radioactivity monitoring equipment will not be affected by this postulated incident, and will be available to the operator.

7.3.2.2 System Evaluation

Redundant instrumentation has been provided for all inputs to the protective systems and vital control circuits.

Where wide process variable ranges and precise control are required, both wide range and narrow range instrumentation is provided.

Instrumentation components were selected from standard commercially available products with proven operating reliability.

All electrical and electronic instrumentation required for safe and reliable operation is supplied from the vital instrumentation buses.

7.3.2.2.1 Pressurizer Pressure

Any accident condition requiring emergency core cooling would involve low pressurizer pressure. The present design for emergency core cooling is accomplished by the SIS actuation from primary system variables. Actuation is initiated by low pressurizer pressure with 2/3 logic. This coincidence arrangement will prevent false actuation of the SIS in the event of a spurious pressure signal. A safety injection block switch is provided to permit the primary system to be depressurized and its water level lowered for maintenance and refueling operation without actuation of the SIS.

This manual block switch is interlocked with pressurizer pressure in such a way that the blocking action is automatically removed above a preset pressure as operating pressure is approached. If two-out-of-three pressure signals are above this preset pressure, blocking action cannot be initiated. The block condition is annunciated in the Control Room.

7.3.2.2.2 SG Level Control During Plant Cooldown

The successful operation of the ESF involves only actuation control functions, with one exception. This exception is the SG level control function associated with plant cooldown using the auxiliary feedwater pumps. This level control system involves remote manual positioning of feedwater flow control valves in order to maintain proper SG water level. SG water level indication and controls are located in the Control Room and at a local control station.

7.3.2.2.3 Motor and Valve Control

For starting pump and fan motors, the control relays, when deenergized, cause the closing coil on the motor starter or circuit breaker to be energized through redundant sets of contacts. When motor starters are used, the starter operating coil is supplied by power from the same source as the subject motor. When circuit breakers are used for motor control, the circuit breaker close and trip coils are supplied by power from a 125 V DC battery bus as outlined in Section 8.3.

For valve motor control, the control relay causes the coil on the main contactor for the closing circuit to be energized. The closing circuit is deenergized by the torque switch on the valve operator, thereby ensuring that the valves have closed to a leak-tight position. Air actuated containment isolation valves are spring-loaded to close upon loss of air pressure.

7.3.2.3 Function Initiation Period

The ESF instrumentation equipment inside the containment is designed to operate under the accident environment of a steam-air mixture and radiation. Environmental design is discussed in Section 3.11.

7.3.2.4 Testing and Reset Prevention

The ESF are designed so that once actuated they remain in the emergency mode upon reset or removal of the ESF actuation signal. The only ESF signal that can be over-ridden is safety injection after a two minute programmed delay. This override condition is indicated by status lights and annunciation in the Control Room.

7.4.1.2.7 Emergency DC Power Supply/Dedicated Shutdown (DS) Instrumentation

The modification provides an alternate DC control voltage source to the emergency switchgear. Figure 7.4.1-6 depicts the alternate supply transfer and emergency switchgear alignment. The new distribution panel B provides an alternate DC control voltage source to 4.16 kV buses 3 and 4 via interconnections with bus 3 cubicle 21 and to 480V buses 2B and 3 via interconnections with bus 2B cubicle 12B. The new distribution panel A provides an alternate DC control voltage source to 4.16 kV buses 1 and 2 via interconnections with bus 1 cubicle 7 and to 480V buses 1 and 2A via interconnections with bus 1 cubicle 1. The transfer to the alternate 125V DC control voltage source is annunciated in the Control Room. In addition, the new 125V DC panel A supports a 5 kVa inverter located outside of the emergency switchgear room, in the 4 kV switchgear room. The inverter supports DS instrumentation which is located in the charging pump room control panel. The DS instrumentation provides the following local displays at the charging pump room panel:

a) Primary system hot and cold leg temperatures
b) Condensate storage tank level
c) Nuclear instrumentation
d) Steam generator 1 level
e) Steam generator 2 level
f) Steam generator 3 level
g) Pressurizer level
h) Pressurizer pressure

With the exception of the nuclear instrumentation, duplicate shutdown instrumentation displays are provided at the turbine mezzanine control panel. Also, the condensate storage tank level is displayed.

The new DC panel A also provides 125V DC at the turbine mezzanine control panel. The 125V DC at the turbine mezzanine panel is not required for the shutdown modification and is available for future plant use.

It should be noted that the emergency DC power configuration shown on Figure 7.4.1-6 is based on the assumed availability of offsite power. In the event of a fire in the battery room (or cable spreading room) coincident with a loss of offsite power, a complete loss of site DC power may be experienced. The main consequence of this failure would be the inability to operate circuit breakers to line up alternate power sources (most breakers are DC-operated).

7.4.1.2.8 Breaker Controls

Local controls have been provided for all circuit breakers required for supply of power to shutdown related switchgear. These controls consist of the following for each affected breaker:

a) Control transfer switch, which functionally disables and isolates the breaker remote control circuits.

b) Selector switch, electrically functional only when the control transfer switch is in the "local" position. This switch provides breaker close and trip control.

7.4.1.2.9 Separation of Power and Control Cables

The existing design is in compliance with the intent of the separation criteria defined by Regulatory Guide 1.75. The isolation criteria applied in this design are consistent with those of the existing plant designs and with the requirements of IEEE-279 and IEEE-384-1977.

All new components that provide alternate power or control capabilities for the DS system have been located so that alternate power sources or control stations will not be affected by any fire that could damage the normal shutdown systems. In addition, all conduits and cable for the DS systems have been routed through areas that will not be affected by fires that could damage systems normally required for shutdown.

In some cases existing power and control cables for the normal shutdown equipment have been rerouted to avoid hazardous areas. All new cable has been installed in rigid steel conduit routed through areas remote from cables presently used for the normal shutdown systems.

7.7.1.5.4 Fixed Incore Detectors

As an optional method for obtaining data on incore flux distribution, four strings of fixed incore detectors are provided. Each fixed incore string has four detectors located at approximately 1/8, 3/8, 5/8, and 7/8 of the distance from the bottom to the top of the core.

7.7.1.6 Axial Power Distribution Monitoring System

The purpose of the Axial Power Distribution Monitoring System (APDMS) is to provide periodic surveillance of the axial peaking factor (F_z) when the reactor is operating at core power levels requiring F_z monitoring. The APDMS is a surveillance and alarm system only and performs no operational plant control or protection functions.

The APDMS is dependent upon the Flux Mapping System for initial selection and positioning of detectors, and for some selected functions during APDMS operation. Four detector/drive assemblies of the Flux Mapping System are utilized by the APDMS. The four particular thimbles and detectors to be used will be selected at the Flux Mapping System control console and the detectors positioned at their parked position inside the reactor vessel, but below the core. With this setup completed, control of the detector drives will be switched to the APDMS mode. Conversely, normal use of the Flux Mapping System requires that the APDMS be deactivated.

There are four control board annunciators:

a) "APDMS High F_z"

b) "APDMS Malfunction"

c) "APDMS on Test"

d) "APDMS not in Normal Mode"

The APDMS is automatically activated into normal operation when reactor power is at or above a preset power level as indicated by either of two Nuclear Instrumentation System power range signals and a single scan is made. The APDMS is automatically deactivated when reactor power is below the preset power level. The power setpoint will be fixed, but manually adjustable. While the system is active, a scanning sequence will be automatically initiated by significant movement of Control Bank D full length rods. The rod bank demand signals are monitored and integrated such that cumulative full length demanded rod motion (in either direction) outside a preset deadband will initiate an automatic typical scanning sequence which consists of scans at preset times following initiation.

In the event no rod movement occurs beyond the preset deadband, a scan will be automatically initiated if the time since the last scan exceeds eight hours. A manual scan can be initiated at any time, except when a scan is already in progress, without disruption of normal operation. When a scan is called for, two detectors are inserted at slow speed (12 ft/min). The F_z calculations begin at bottom-of-core and end at top-of-core. Both detectors are then

withdrawn to their park positions. The calculated value from each detector F_z computer and the Time of Scan will be displayed on the APDMS Scan Display panel. The readouts will be retained until the next APDMS scan is performed, at which time the readouts will change to display the updated information. Since F_z is a simple calculation of the peak to average factor for the detector trace, it is not necessary to normalize or cross-calibrate the movable detectors at part of APDMS operation.

For xenon ocsillations caused by changes in power level, and/or rod movements, plant procedures exist which provide for operator initiated damping of the oscillations by control rod movement. Adverse axial power distributions caused by xenon shifts which result from routine load changes during power operation are controlled using Constant Axial Offset Control (CAOC) procedures. The CAOC procedure limits the peaking factor to the Technical Specification limit by restricting xenon redistribution during power changes. This is done by monitoring the power difference between the top and bottom of the core as a function of different power levels and core conditions.

7.7.1.7 Automatic Load Dispatch

Load changes on generating units in the Carolina Power and Light Company (CP&L) System are initiated by a computer located at a central system dispatch center. This computer constantly receives information from the system on the load requirements, compares this information to the generation on the system, and automatically sends out signals to the generating plants to adjust the plant generation to match the load requirements. Generation is allocated to the plants in such a manner that the incremental cost of the power delivered to the load by all units is equal, taking into account fuel costs and transmission losses.

Experience has proved this method of load dispatch to be very satisfactory from a generating plant standpoint as well as from a system operation standpoint.

From a generating plant standpoint, the following three criteria have been strictly adhered to in order to ensure that operation of all plants are maintained well within the bounds of what is considered good operating practice from a safety standpoint as well as from an equipment reliability standpoint.

a) The plant control operator, by operating one switch located on the plant control board, has the ability to switch the plant control system to a mode of operation that will make the plant unresponsive to load change signals from the load dispatch computer.

It is the responsiblity of the plant control operator to switch to this mode of operation at any time if in his judgment the generation of his plant should not be changed due to some condition within the plant.

b) The plant control operator, by the adjustment of two dials located on the plant control board, has the ability to set the generation range over which the plant will respond to signals to increase or decrese generation on the plant.

If there is a requirement for ESF operation coincident with undervoltage on the 480 volt bus, the ESF equipment is sequentially started as shown in Table 8.3.1-5.

Motor control centers are energized upon closing of the generator breaker and injection valves are opened.

In the event one emergency generator does not come on the line, the tie bus connection to the energized diesel bus is automatically closed permitting the start of the second SI pump.

Should any of the feeder breakers associated with the safety features components or the 480 volt bus tie breaker trip due to overload, the trip is indicated in the Control Room. The breakers can be manually reclosed from the Control Room. Overload trip elements on the reversing starters associated with the various motor-operated valves can be reset at the motor control centers.

8.3.1.1.5.4 Test and Inspection Capabilities

The diesel generators are tested to assure that they will provide power for operation of equipment. These tests also assure that the emergency system controls and the control systems for safety features equipment will function automatically in the event of a loss of all normal 480 V AC station service power. The starting of the diesel generator sets can be tested from their respective rooms.

The testing frequency is often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The control components are in dust-tight enclosures. The fuel supply and starting circuits and controls are continuously monitored and faults are alarm indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

To verify that the emergency power system will respond within the required time limit and properly when required, surveillance testing is conducted as required by the Technical Specifications.

8.3.1.2 Analysis

The plant was built before the inception of the various Institute of Electrical and Electronic Engineers (IEEE) standards, Regulatory Guides, and other criteria now in place. The system was however compared to General Design Criteria 2 and 39, as discussed in Section 3.1.

8.3.1.2.1 Studies

Several studies of plant electrical system adequacy have been done. A study of degraded grid voltage effects on the plant is given in References 8.3.1-1 and 8.3.1-2.

A study of the requirements for degraded grid voltage led to the generation of the "System Design Basis for Degraded Grid Voltage and Emergency Power System Modification" (Reference 8.3.1-3).

A study of equipment needed to safely shut down the plant in the event of a loss of the E1, E2 buses or cable spreading room was done as part of the fire protection report (Reference 8.3.1-4).

The degraded grid voltage studies demonstrated that expected plant conditions of .95 per unit to 1.06 per unit were acceptable to running expected safety loads within required voltage tolerances. A second level of voltage protection was added however. This protection installed on the 480 V AC bus will cause a trip of the bus and loading onto the diesel at 86 percent of 480 V AC maintained for 10 sec. The protection circuitry prevents load shedding when the bus is already on the diesel.

8.3.1.2.2 Reliability Assurance

The electrical system equipment is arranged so that no single contingency can inactivate enough safety features equipment to jeopardize the plant safety. The 480 volt equipment is arranged on 8 buses. The 4160 volt equipment is supplied from 4 buses.

Multiple outside sources of power are available to the plant. Normal operations utilize both outside and unit-generated power. Separation of these two sources is maintained in the 4160 volt, 480 volt, and lower voltage systems. See Figure 8.1.2-1.

The plant auxiliary equipment is arranged electrically so that redundant items receive their power from the two different sources. Two charging pumps, four service water pumps, four containment fans, and two residual heat pumps are divided between 480 volt buses E1 and E2. An alternate feed to service water pump D and MCC 5, and a primary feed to component cooling pump A and charging pump A are all supplied from the 480 volt dedicated shutdown bus. Redundant valves are supplied from motor control centers connected to buses E1 and E2.

Refer to Table 8.3.1-5 for the engineered safety features automatic actuation sequence and times after the initiation signal for the cases when the normal power source is available and when only the diesel power source is available.

The components of the sequencing circuits are control relays and electro-pneumatic timing relays. These relays are standard devices universally used in control circuits. One control relay or one timing relay is used to close each circuit breaker feeding 480 volt, 3 phase power to the safety features components. The control power for the relays is supplied from the station batteries. Battery A supplies the sequencing circuits for safety features actuation train A. The sequencing circuits for the two safety features actuation trains are located in separate safety features actuation relay racks in the relay room.

When there is voltage on the associated 480 volts emergency buses (see Figure 8.3.1-3) operation of the master SI relay initiates safety features sequencing circuit by energizing auxiliary control and timing relays. These timing relays actuate the starting sequence as shown in Table 8.3.1-5.

TABLE 8.3.1-1

EMERGENCY LOAD SCHEDULE FOR DIESEL GENERATORS

A. EMERGENCY DIESEL GENERATOR LOADS FOR A LOSS-OF-COOLANT ACCIDENT

COMPONENT	RATED HP/DIESEL	INJECTION PHASE		
		UNITS REQ'D	bhp	kW
Auxiliary Building Charcoal Filter Fan	5	0	0	0
Battery Charger	-	0	0	0
Auxiliary Building Exhaust Fan	75	1	58	48
Auxiliary Building Supply Fan	50	1	42	35
Auxiliary Feedwater Pump	350	1	322	267
Component Cooling Pump	350	0	0	0
Contaiment Fan Cooler	350	2	482	400
Containment Spray Pump	200	1	200	166
Control Room Air Conditioning	15	1	15	12
Diesel Room Ventilation	20	1	20	17
ESF Component Chiller Units (3)	15	3	15	12
Instrument Air Compressor	50	0	0	0
MCC (Valves and Emergency Lighting)	-	1	48	40
MCC (Reactor Auxiliary Building)	-	1	87	72
Residual Heat Removal Pump	300	1	300	249
Safety Injection Pump	350	2	736	611
Seal Oil Pump	20	0	0	0
Service Water Booster Pump	100	1	98	81
Service Water Pump "B" Bus El Only	300	2	580	481
Turning Gear	50	0	0	0
Turning Gear Oil Pump	75	0	0	0
			2903	2329

TABLE 8.3.1-1 (Cont'd)

COMPONENT	RECIRCULATION PHASE: HIGH HEAD RECIRCULATION CASE UNITS REQUIRED	bhp	kW	RECIRCULATION PHASE: LOW HEAD RECIRCULATION CASE UNITS REQUIRED	bhp	kW
Auxiliary Building Charcoal Filter Fan	1	5	4	1	5	4
Battery Charger	1	20	17	1	20	17
Auxiliary Building Exhaust Fans	1	58	48	1	58	48
Auxiliary Building Supply Fans	1	42	35	1	42	35
Auxiliary Feedwater Pump	1	322	267	0	0	0
Component Cooling Pump	1	350	291	1	350	291
Containment Fan Cooler	2	300	249	2	482	400
Containment Spray Pump*	0	0	0	1	200	166
Control Room Air Conditioning	1	15	12	-	15	12
Diesel Room Ventilation	1	20	17	1	20	17
ESF Component Chiller Units	3	15	12	2	15	12
Instrument Air Compressor	1	50	42	1	50	42
MCC (Valves and Emergency Lighting)	1	48	40	1	48	40
MCC (Reactor Auxiliaries)	1	87	72	1	87	72
Pressurizer Heaters**	3	183	150	3	183	150
Residual Heat Removal Pump	1	300	249	1	300	249
Safety Injection Pump	1	368	305	0	0	0
Seal Oil Pump	1	0	0	1	20	17
Service Water Booster Pump	1	98	81	1	98	81
Service Water Pump "B" Bus E1 Only	1	580	481	1	580	481
Turning Gear	1	50	42	1	50	42
Turning Gear Oil Pump	1	75	63	1	76	63
		2803	2327		2516	2089

*For the high head recirculation case in the event the containment pressure rise is high enough to initiate containment spray the containment spray pump would run increasing the diesel load to 2493 kW.

**Pressurizer heaters are manually loaded on buses. These will not be added until approximately one hour into the event. These loads are split between buses. They are shed upon initiation of a safety injection signal and will not be loaded onto the bus when the safety injection pumps are running.

TABLE 8.3.1-1 (Cont'd)

B. LOAD SCHEDULE FOR DIESEL GENERATORS FOR LOSS OF OFFSITE POWER (BLACKOUT SEQUENCE)

	UNITS REQ'D	RATED HP	bhp	kW
Auxiliary Feedwater Pump	1	350	322	267
Service Water Pump	1	300	290	240
Service Water Booster Pump	1	100	98	81
Component Cooling Pump	1	350	350	290
Motor Control Centers				
Turning Gear	1	50	50	42
Critical Valves and Emergency Lighting	-	48	48	40
Diesel Fuel Oil Pump	1	0.5	0.5	0.4
Communications	-	-	-	2
Boric Acid Tank Heater	1	-	-	15
Instrument Bus	1	-	-	7.5
Heat Tracing	1	-	-	45
Control Room Air Conditioning	1	15	15	12
Diesel Room Ventilation	1	20	20	17
Auxiliary Feedwater Pump Chiller Unit	1	9	9	7.5
Diesel Auxiliaries	1	-	-	2.6
Boron Injection Tank Heaters	1	-	-	7.5
Total				1076

8.3.3 FIRE PROTECTION FOR CABLE SYSTEMS

Cable loading of trays and, consequently, heat dissipation of cable throughout the plant have been carefully studied and controlled to ensure no overloading. The criteria for electrical loading were developed using Insulated Power Cable Engineers Association (IPCEA) Standard P-46-426, Manufacturer Recommendations and Good Engineering Practice.

Derating factors for cables in trays without maintained spacing were taken from Table VIII of the IPCEA publication. Derating factors for the maximum ambient temperature existing in any area of the plant were also taken from the IPCEA publication. These factors were applied against ampacities selected from appropriate tables in other portions of the standard.

For physical loading of trays, the following criteria were followed: 4 kV power, one horizontal row of cables was allowed in a tray; 480 volt power, 30 percent of the cross-sectional area of the tray is filled for control and instrumentation; 70 percent of the cross-section of a tray was the maximum fill. This was exceeded in 6 cases which have been analyzed and found to be satisfactory (Reference 8.3.3-1).

In general, for instrumentation cables, four basic channels are routed through the plant. These channels include cables for systems 65 volts and less. Cables assigned to these four channels will remain in their respective channels throughout the run.

Certain other cables are run in with the four instrument channels; such as, thermocouple cable, public address system cabling and instrument power supplies.

To assure that only fire retardant cables are used throughout the plant, a careful study of cable insulation systems was previously undertaken.

Insulation systems that appeared to have superior flame retardant capability were selected. An extensive flame testing program took place which included ASTM vertical flame testing and bonfire tests. Cables were specified on the basis of results from these tests.

The following tests were made to determine the flame resistant qualities of various cable coverings and insulations:

a) Standard Vertical Flame Test - made in accordance with ASTM-D-470-59T, Tests for Rubber and Thermalplastic Insulated Wire and Cable

b) Five Minute Vertical Flame Test - made with cable held in vertical position and 1750^{o}F flame applied for 5 minutes, and

c) Bon-Fire Test - consisting of exposing, for 5 minutes, bundles of three or six cables to flame produced by igniting transformer oil in a 12 in. pail. The cable was supported horizontally over the center of the pail, the lowest cable located 3 in. above the top of the pail. The time to ignite the cable and the time the cable continued to flame after the fire was extinguished were noted.

To prevent spread of flame, fire barriers of glass wool packed around trays on approximately 10 ft spacing in vertical runs have been provided. Fire barriers of similar construction are used in all trays when they penetrate walls.

Detectors which are sensitive to temperature and the rate of change of temperature are provided for fire detection and alarm in unattended areas where large concentrations of cables are installed.

In areas where missile protection could not be provided, such as near the reactor coolant system (RCS), redundant instrument impulse lines and cables were run by separate routes. These lines were kept as far apart as physically possible, or were protected by heavy (1/4 in.) metal plates interposed where inherent missile protection could not be provided by spacing.

Cable trays are entirely of metal construction and present no combustible hazard. Safety-related cable trays outside the cable spread room have been evaluated for fire protection provisions.

Since safety-related cable runs in the Auxiliary Building do not satisfy the requirements of Regulatory Guide 1.75 and consist primarily of polyvinyl chloride (PVC) jacketed cables, a flame retardant coating was applied for trays containing engineered safeguards cable. Cabling inside containment has silicone rubber jacket material which has superior fire resistant properties compared to the PVC cable. Automatic water sprinklers were installed to protect safety-related cable in the hallway of the Auxiliary Building ground floor near the station air compressors. No critical equipment requiring protection from water damage is in the area. This and other areas have manual hose stations and fire extinguishers for additional protection. A cable tray fire would be a rather slowly propagating fire (1-2 in./min) even without flame retardant coating, so use of a coating and automatic detection would provide adequate protection unless there is a significant exposure fire hazard.

Cable and cable tray penetrations of fire barriers have been sealed to give adequate fire resistance. Cables which enter the cable spreading room from the electrical equipment room do so via trays which pass through openings cast in the wall just large enough to allow tray passage. The air spaces around the trays and cables are sealed by field fitted aluminum plates, glass wool and Flamemastic protective coating. Control wiring entering the Control Room from the cable spreading room uses slots in the floor. Each slot is sealed with a plate fabricated and drilled so that cables pass through individual holes in the plate, the cables are individually sealed and supported. In other locations where electrical cable trays penetrate the walls and floors, they are packed with glass wool and protective Flamemastic coating.

Sketches describing cable penetrations for walls and floors are given in Figures 8.3.3-1 and 8.3.3-2. The penetration design is similar to that recommended in the International Guidelines for Fire Protection of Nuclear Power Plants (1974).

CHAPTER 9
AUXILIARY SYSTEMS

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CHAPTER 9

AUXILIARY SYSTEMS

LIST OF TABLES

All valves in the SWS were hydrostatically tested at three times the design pressure on the body and two and one-third times design pressure on the seat. SWS design pressure is 150 psig.

All service water piping, except for the piping downstream of the service water booster pumps, was hydrostatically tested in the field at 225 psig or one and one-half times design. Service water piping downstream from the service water booster pumps was hydrostatically tested in the field at 150 psig or one and one-half times design. The welds in shop-fabricated service water piping were liquid penetrant or magnetic particle inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. The Inservice Inspection Program for the SWS is contained in Section 3.9.

Electrical components of the SWS can be tested periodically.

9.2.5 CONDENSATE STORAGE FACILITIES

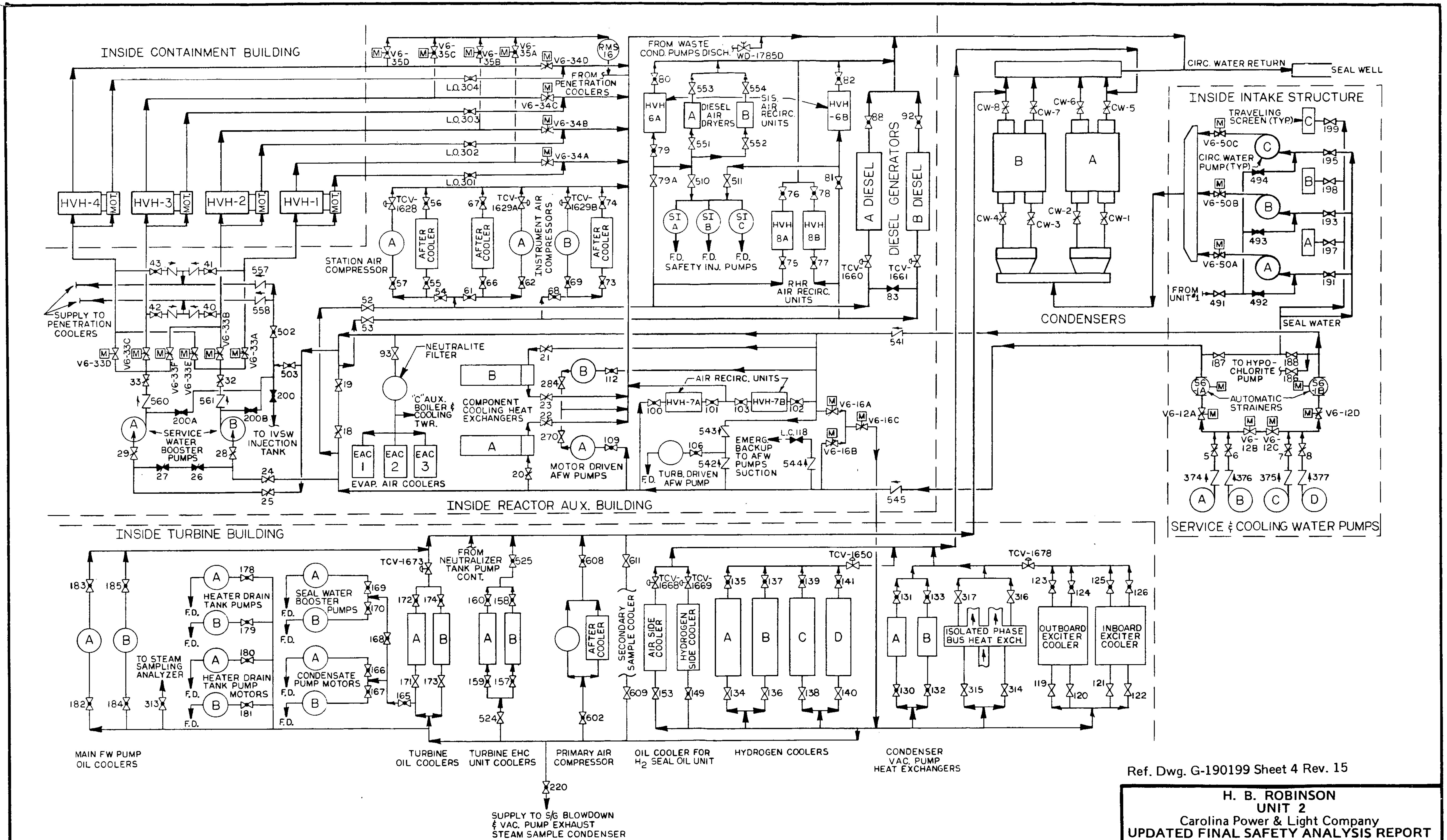
The Condensate Storage Tank is a 200,000 gal capacity tank constructed of stainless steel per ASTM alloy type 304. The tank is Seismic Class I designed under the American Water Works Association Standard for Steel Tanks Code (AWWA D100).

9.2.5.1 Safety Evaluation

If complete failure of the condensate storage tank occurs due to tornado damage, the water source for auxiliary feed pumps can be changed to either the service water system or the deep well pumps. Thus, safe shutdown and cooldown can be assured.

The Technical Specifications require a minimum of 35,000 gal of water (approximately 19 percent indicated level) in the condensate storage tank for normal make-up to the secondary system. An unlimited backup water supply is available from the Robinson Impoundment via the plant Service Water System. Although not condensate grade, the service water is of sufficient quality to provide long term cooling to the steam generators without adverse affects. The 35,000 gal requirement satisfies the amount needed for at least two hours of operation in hot standby conditions following a complete loss of turbine-generator and offsite electrical power. The tank level is normally maintained as full as possible.

Piping associated with the condensate storage tank is either buried or is routed in a trench which precludes exposure to the effects of a tornado.



Ref. Dwg. G-190199 Sheet 4 Rev. 15

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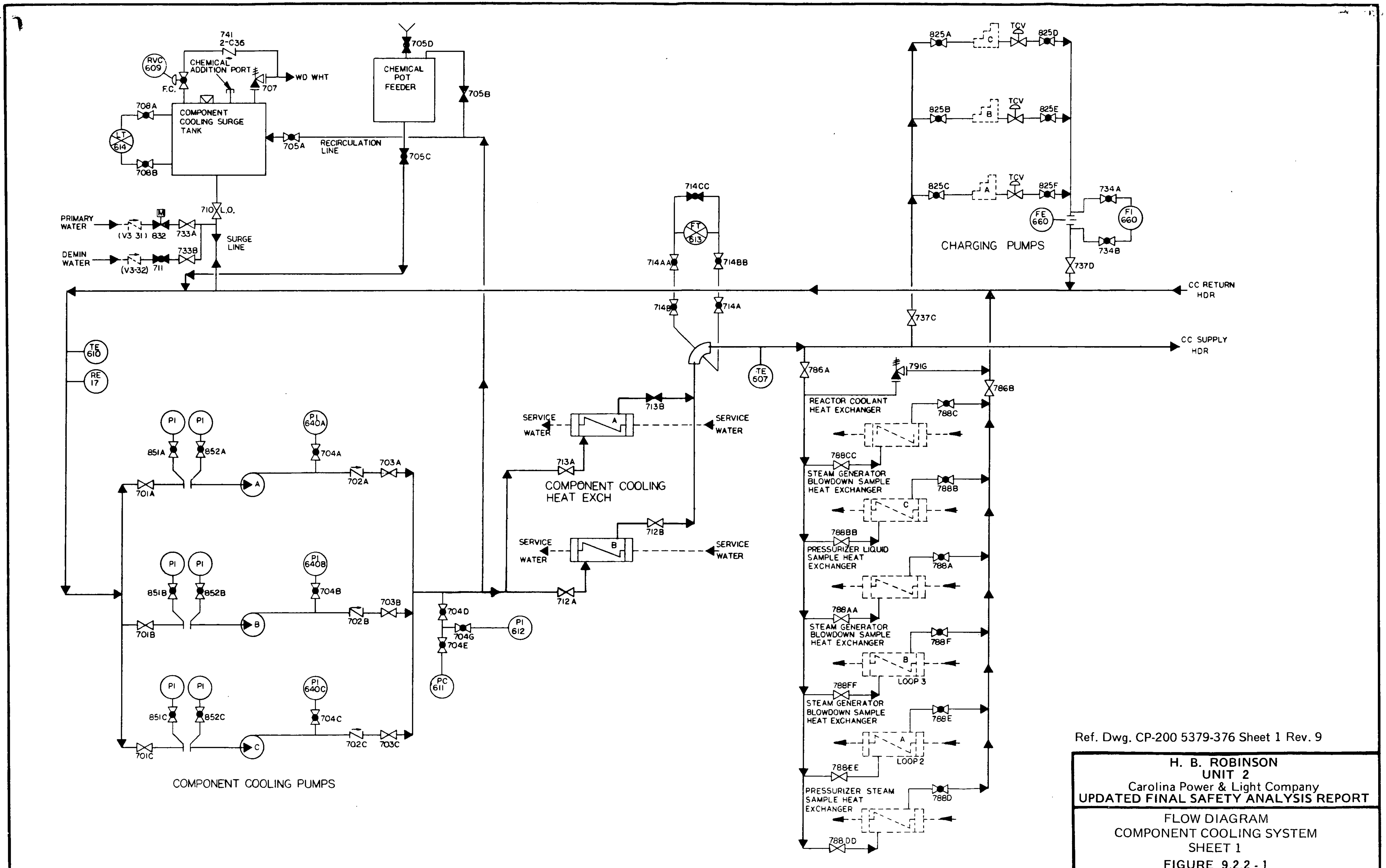
FLOW DIAGRAM
SERVICE AND COOLING WATER SYSTEM

FIGURE 9.2.1 - 1



Ref. Dwg. CP-200 5379-376 Sheet 1 Rev. 9

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FLOW DIAGRAM
COMPONENT COOLING SYSTEM
SHEET 1
FIGURE 9.2.2 - 1

9.3.2 PROCESS SAMPLING SYSTEM

This system provides samples for laboratory analysis to evaluate reactor coolant, and other reactor auxiliary systems chemistry during normal operation.

9.3.2.1 Design Basis

The design basis for the Sampling System are:

a) The system is capable of providing reactor coolant samples during both normal reactor operating conditions and cooldown when the system pressure is low and the residual heat removal loop is in operation

b) Access to the containment is not a requirement

c) Sampling of other process coolants, such as tanks in the waste Disposal System, can be accomplished locally, and

d) Equipment for sampling secondary and nonradioactive fluids is separated from the equipment provided for reactor coolant samples.

The system component code requirements are given in Section 3.2.

9.3.2.2 System Description

9.3.2.2.1 General Description

The Sampling System, shown in Figure 9.3.2-1, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the Reactor Coolant, Auxiliary Coolant, Steam, and Chemical and Volume Control Systems (CVCS). Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentration adjustments, evaluating fuel element integrity and mixed bed demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The Sampling System is designed to be operated manually, on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown.

Reactor coolant liquid lines, which are normally inaccessible and require frequent sampling, are sampled by means of permanently installed tubing leading to the sampling room.

Sampling System equipment is located inside the Auxiliary Building with most of it in the sampling room. The delay coil and sample lines with remotely operated valves are located inside the reactor containment.

Reactor coolant hot leg liquid, accumulator liquid, pressurizer liquid and pressurizer steam samples originating inside the reactor containment flow through separate sample lines to the sampling room. Each of these connections

to the Reactor Coolant System (RCS) has a remotely operated isolation valve located close to the sample source. The samples pass through the reactor containment compartment, to the Auxiliary Building, and into the sampling room, where they are cooled (pressurizer steam samples are condensed and cooled) in the sample heat exchangers. The sample stream pressure is reduced by a manual throttling valve located downstream of each sample pressure vessel. The sample stream is purged to the volume control tank in the CVCS until sufficient purge volume has passed to permit collection of a representative sample. After sufficient purging, the sample pressure vessel is isolated and then disconnected for laboratory analysis of the contents.

Alternately, liquid samples may be collected by bypassing the sample pressure vessels. After sufficient purge volume has passed to permit collection of a representative sample, a portion of the sample flow is diverted to the sample sink where the sample is collected.

The reactor coolant sample originating from the residual heat removal loop of the Auxiliary Coolant System has a remotely operated, normally closed isolation valve located close to the sample source. The sample line from this source is connected into the sample line coming from the hot leg at a point ahead of the sample heat exchanger. Samples from this source can be collected either in the sample pressure vessel or at the sample sink as with hot leg samples.

Liquid samples originating at the CVCS letdown line at demineralizer inlet and outlet pass directly through the purge line to the volume control tank. Samples are obtained by diverting a portion of the flow to the sample sink. If the pressure is low in the letdown line, the purge flow is directed to the chemical drain tank. The sample line from the gas space of the volume control tank delivers gas samples to the volume control tank sample pressure vessel in the sampling room. Purge flow for these samples is discharged to the vent header in the Waste Disposal System.

Because samples from the pressurizer steam phase, the reactor coolant dissolved gas and volume control tank gas phase may contain accumulated radioactive gases, the respective sample vessel stations are located in small, well ventilated and shielded cubicles within the sampling room.

Samples of the steam generator liquid are obtained from the blowdown lines. These sample lines are routed by separate lines from each steam generator blowdown line into the sample room. These lines are equipped with two remotely operated isolation valves in each line immediately outside the containment. These valves are automatically closed upon receipt of a signal from the blowdown sample radiation monitor or the containment isolation system.

The sample lines are routed to the sample room where the liquid is cooled and the pressure reduced. Each individual sample is then split into two routes: one goes to the sample sink to provide periodic samples for chemical analysis, and the second goes to a conductivity cell, a radiation monitor and then to the blowdown flash tank. This second line handles a continuous flow for a constant reading of conductivity and a constant monitoring for radiation.

The sample sink, which is contained in the laboratory bench as a part of the sampling hood, contains a drain line to the Waste Disposal System.

Two types of samples are obtained by the system: high temperature-high pressure RCS samples which originate inside the reactor containment, and low temperature-low pressure samples from the Chemical and Volume Control and Auxiliary Coolant Systems. These samples are taken as follows:

a) High Pressure - High Temperature Samples: A sample connection is provided from each of the following high pressure-high temperature samples:

1) The pressurizer steam space
2) The pressurizer liquid space
3) Hot legs of loops 2 and 3
4) Blowdown lines from each steam generator

b) Low Pressure - Low Temperature Samples: A sample connection is provided from each of the following low pressure-low temperature samples:

1) The mixed bed demineralizer inlet header
2) The mixed bed demineralizer outlet header
3) The residual heat removal loop, just downstream of the heat exchangers
4) The volume control tank gas space
5) The accumulators

The high pressure, high temperature samples and the residual heat removal loop samples leaving the sample heat exchangers are held to a temperature of 130°F to minimize the generation of radioactive aerosols.

9.3.2.2.2 Components

A summary of principal component data is given is Table 9.3.2-1.

9.3.2.2.2.1 Sample Heat Exchangers

Six sample heat exchangers reduce the temperature of samples from the pressurizer steam space, pressurizer liquid space, each steam generator and the reactor coolant to 130°F before samples reach the sample vesssels and sample sink. The tubes of the heat exchangers are austenitic stainless steel, and the shells are carbon steel.

The inlet and outlet tube sides have socket-weld joints for connections to the high presure sample lines. Connections to the component cooling water lines are socket-weld joints. The samples flow through the tube side and component cooling water from the Auxiliary Coolant System circulates through the shell side.

9.3.2.2.2.2 Delay Coil

The sample line contains a delay coil, consisting of coiled tubing, which has sufficient length to provide at least a 40 second sample transit time within the containment and an additional 20 seconds transit time from the reactor containment to the sampling hood. This allows for decay of short lived isotopes to a level that permits normal access to the sampling room.

9.3.2.2.2.3 Sample Pressure Vessels

The high pressure sample trains, the residual heat removal loop sample train and the volume control tank gas space sample train each contain sample pressure vessels which are used to obtain liquid or gas samples. The hot leg and the residual heat removal loop sample lines have a single sample pressure vessel in common. Integral isolation valves are furnished with the vesssl and quick disconnect coupling valves containing poppet-type check valves are connected to nipples extending from the valves on each end. The vessels, valves and couplings are austenitic stainless steel.

9.3.2.2.2.4 Sample Sink

The sample sink is located in a hooded enclosure which is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and storage for radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid.

9.3.2.3 Safety Evaluation

The Sampling System is not required for safe shutdown nor to mitigate the consequences of an accident and is therefore designated as a non-safety related system. However, since the system does contain potentially radioactive material, the following discussion indicates the precautions taken to insure safe operation of the plant.

Isolation valves are provided outside the reactor containment which trip closed upon actuation of the containment isolation signal.

The system operates on an intermittent basis, and under administrative manual control.

Leakage of radioactive reactor coolant from this system within the containment is evaporated to the containment atmosphere and removed by the cooling coils of the Containment Air Recirculation and Cooling System. Leakage of radioactive material from the most likely places outside the containment is collected by placing the entire sampling station under a hood provided with an offgas vent to waste gas processing. Liquid leakage from the valves in the hood is drained to the chemical drain tank.

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident (LOCA), and the consequences were analyzed. The results are presented in Table 9.3.2-2. From this evaluation it is concluded that proper consideration has been given to station safety in the system.

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service. With the exception of the sample vessel quick-disconnect couplings and compression fittings at the sample sink, socket welded joints are used throughout the Sampling System. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

Remotely-operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual stop valves are provided for component isolation and flow path control at all normally accessible Sampling System locations. All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material. Manual throttle valves are provided to adjust the sample flow rate as indicated on Figure 9.3.2-1.

Check valves prevent gross reverse flow of gas from the volume control tank (VCT) into the sample sink.

9.3.2.4 Testing and Inspection

Since the Sampling System is in use during plant operation, no special testing or inspection is required.

9.3.2.5 Instrumentation Requirements

The liquid sample flow indicator (FI-903) indicates the sample line flow rate when purging the sample lines to the VCT. The normal rate is 0.42 gpm.

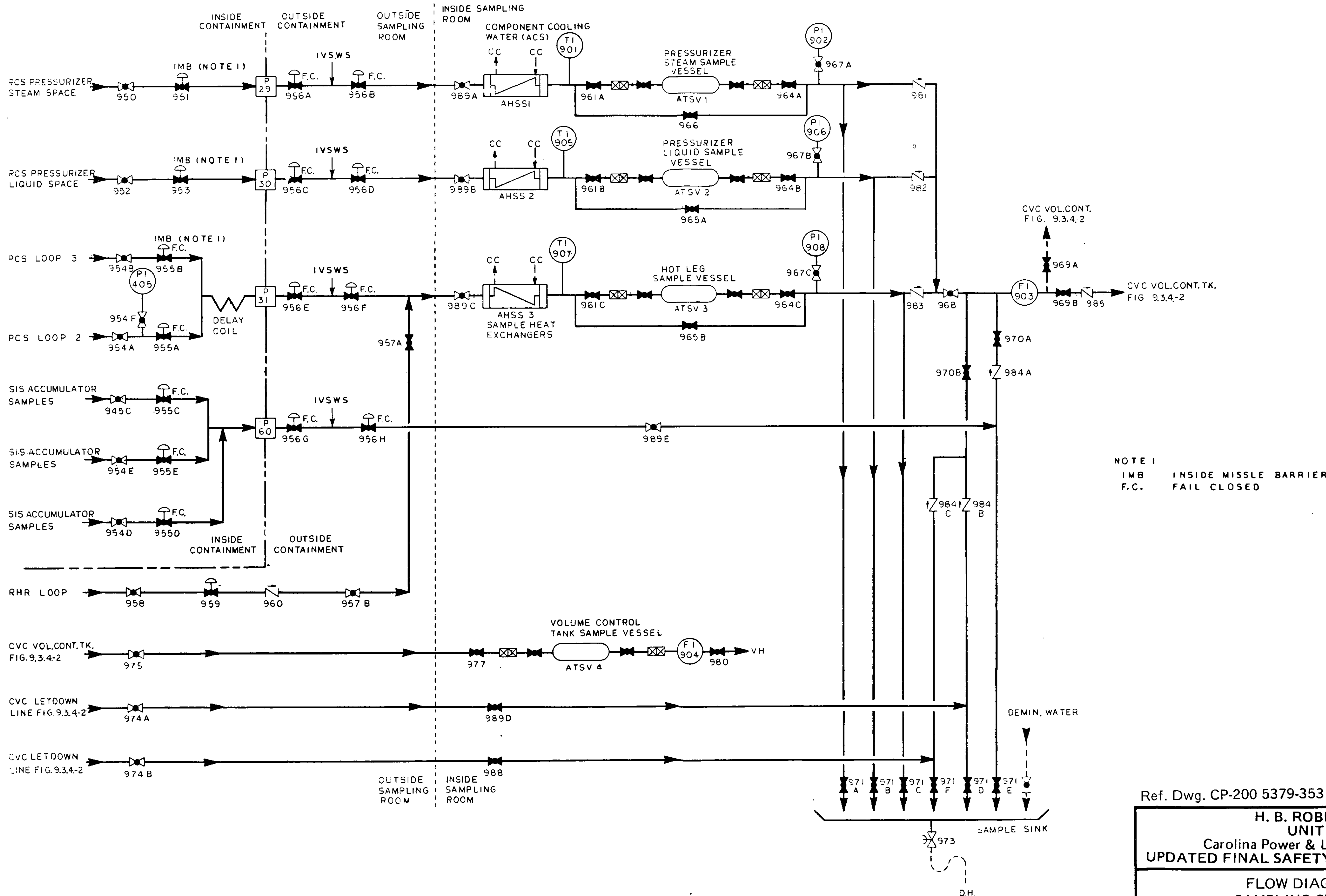
The gas sample flow indicator (FI-904) from the VCT gas space indicates purging flow to the vent header. The normal flow through this line is 1 cfm.

There are local pressure gauges (PI-902, 906, 908) on the sample lines from the pressurizer steam space, pressurizer liquid space, and the hot leg sample line. The pressure in these lines is regulated to 75 psig or less.

There are local temperature indicators (TI-901, 905, 907) on the sample lines from the pressurizer steam space, the pressurizer liquid space and the hot leg sample line. The flow in these lines is throttled so that the sample temperature out of the heat exchanger is 130^{o}F or less.

There are switches for all remotely-operated valves. Valve position indicating lights are also on this panel, located above the switch for the valve.

The valves used for containment isolation purposes, 2 valves per line, are operated by a single switch for both valves.



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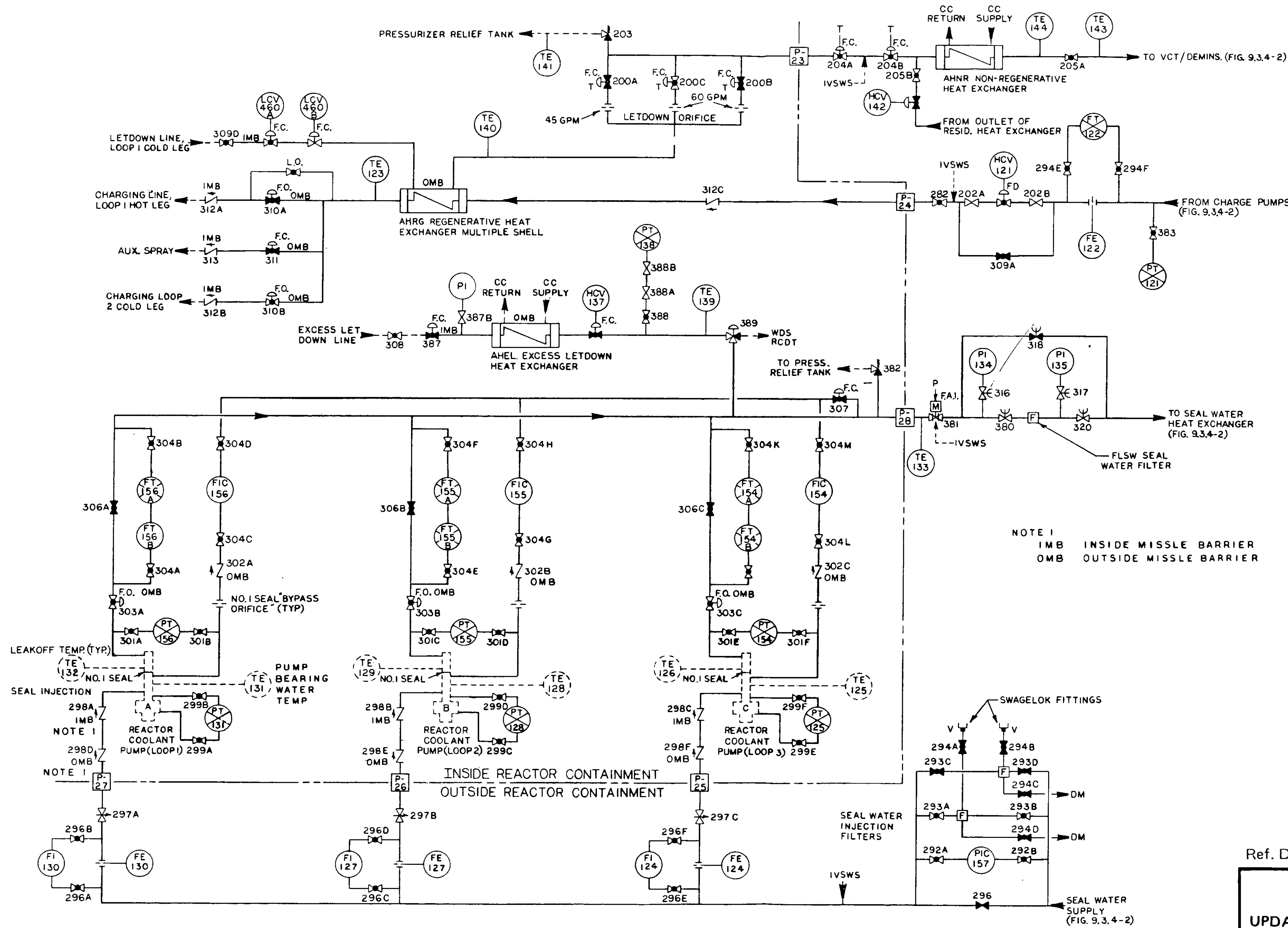
FLOW DIAGRAM
SAMPLING SYSTEM

FIGURE 9.3.2 - 1



Ref. Dwg. CP-200 5379-685 Sheet 1 Rev. 11

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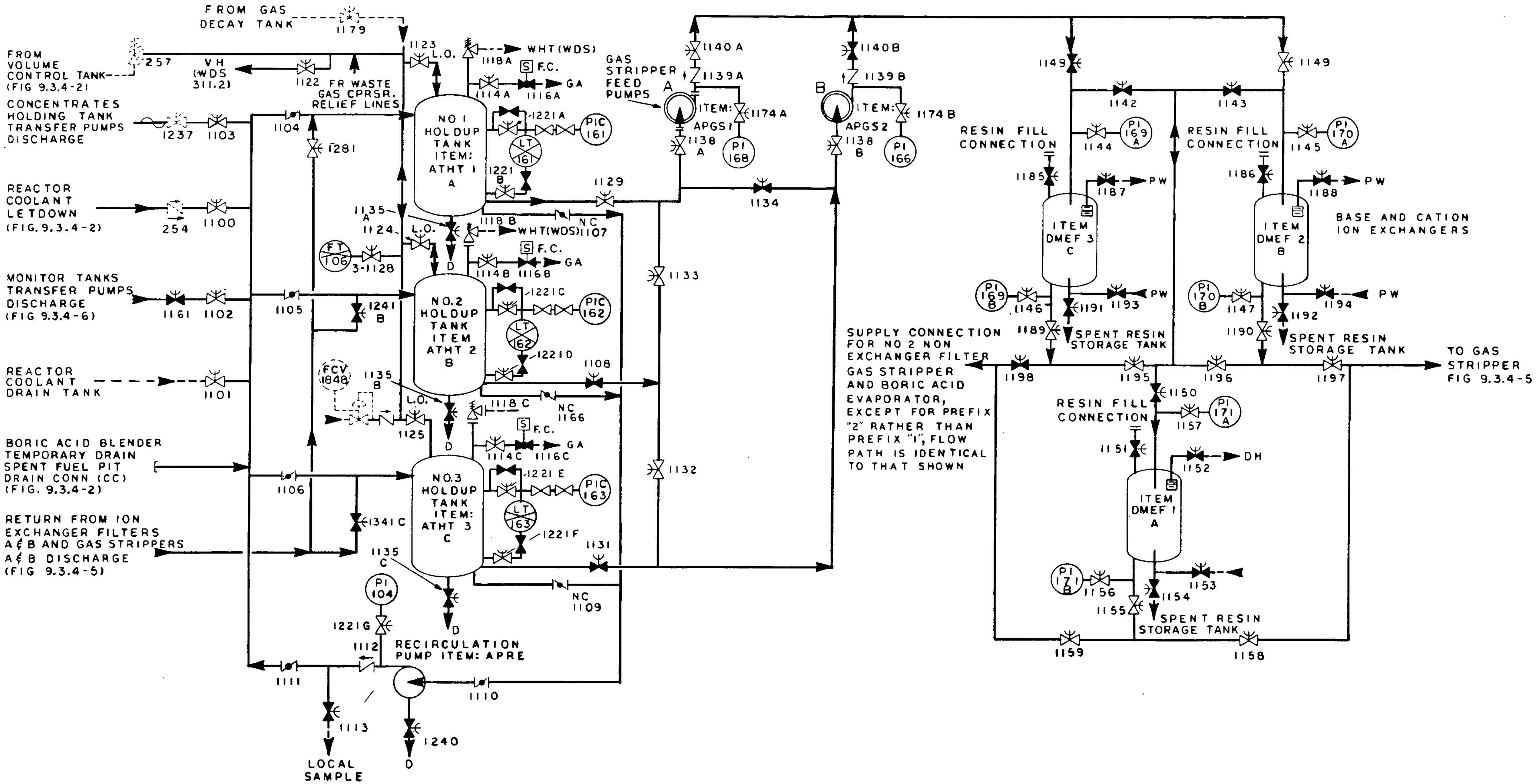
FLOW DIAGRAM
CHEMICAL AND VOLUME
CONTROL SYSTEM
SHEET 1
FIGURE 9.3.4 - 1



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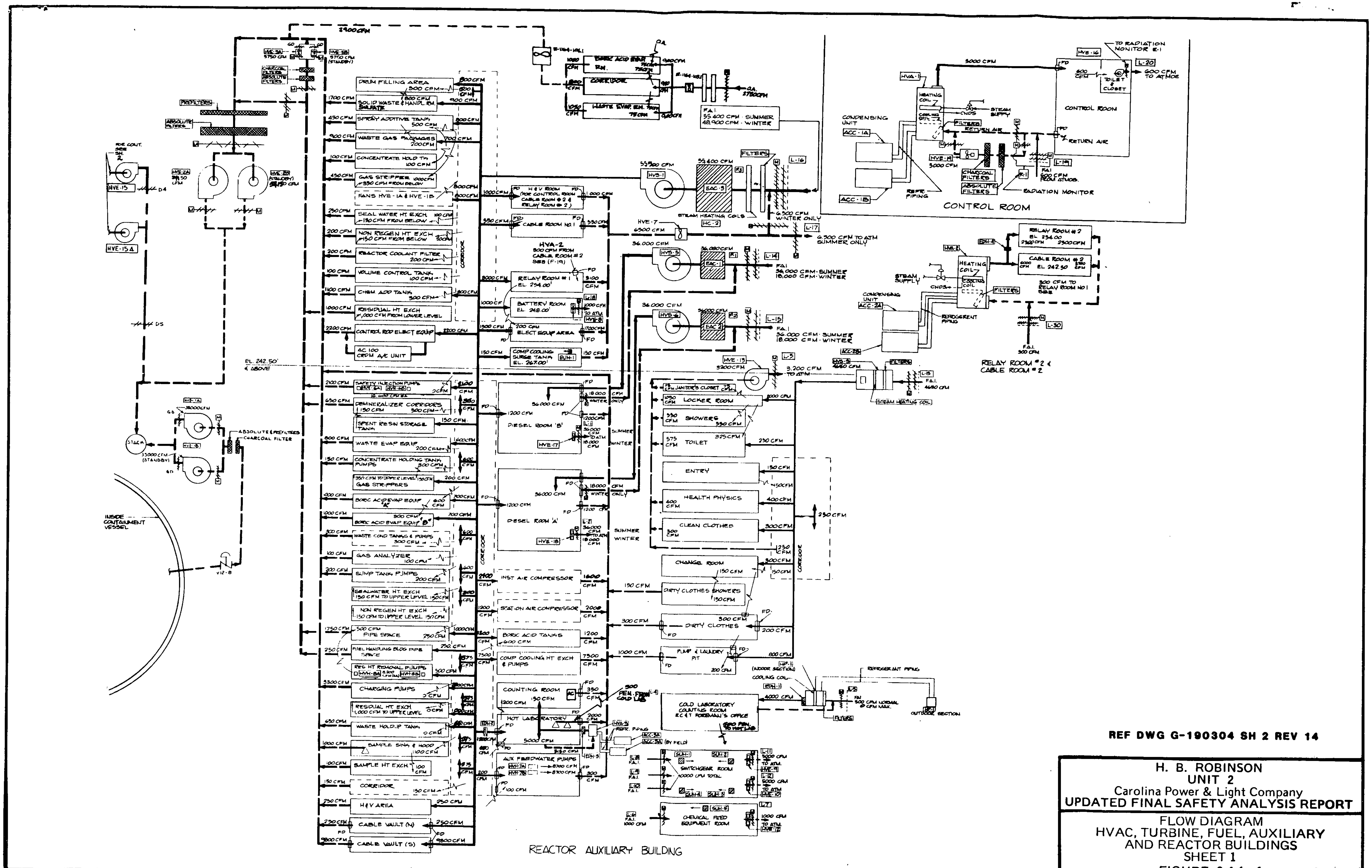
FLOW DIAGRAM
CHEMICAL AND VOLUME
CONTROL SYSTEM
SHEET 4

FIGURE 9.3.4 - 4



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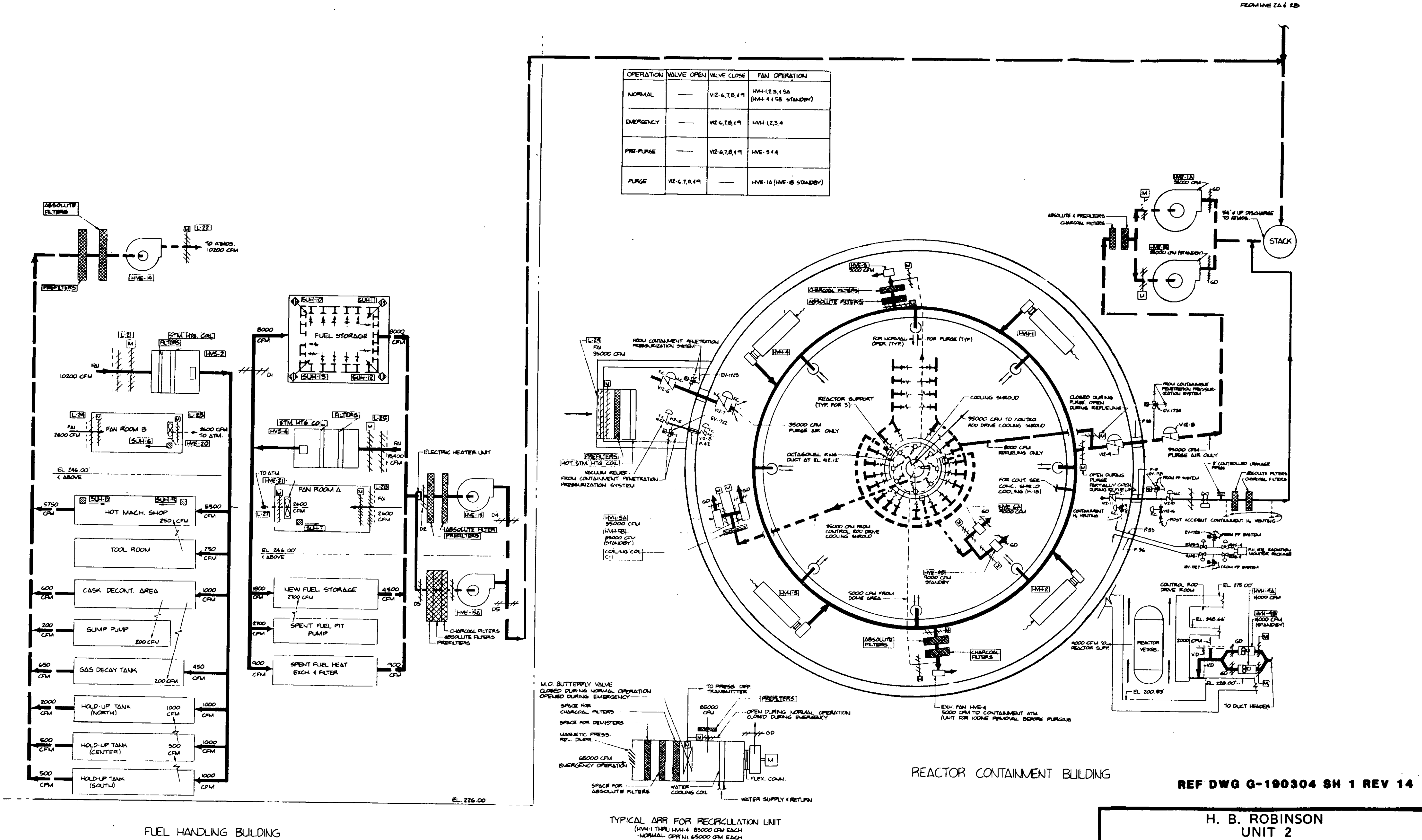
FLOW DIAGRAM
HVAC, TURBINE, FUEL, AUXILIARY
AND REACTOR BUILDINGS
SHEET 1

FIGURE 9.4.1 - 1



OPERATION	VALVE OPEN	VALVE CLOSE	FAN OPERATION
NORMAL	—	V12-6,7,8, & 9	HVH-1,2,3, & 5A (HVH-4 & 5B STANDBY)
EMERGENCY	—	V12-6,7,8, & 9	HVH-1,2,3,4
PRE-PURGE	—	V12-6,7,8, & 9	HVE-5 & 4
PURGE	V12-6,7,8, & 9	—	HVE-1A (HVE-1B STANDBY)

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FLOW DIAGRAM
HVAC, TURBINE, FUEL, AUXILIARY
AND REACTOR BUILDINGS
SHEET 2
FIGURE 9.4.1 - 2

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION SYSTEM

9.5.1.1 Design Bases

The H. B. Robinson Unit 2 (HBR) nuclear plant has been evaluated with regard to fire protection to determine that the total fire protection program provides reasonable assurance that a fire will not cause an undue risk to the health and safety of the public, will not prevent the performance of necessary safe shutdown functions and will not significantly increase the risk of radioactive release to the environment. Appendix A to Branch Technical Position (BTP) Auxiliary and Power Conversion Systems Branch (APCSB) 9.5-1 provides specific guidelines which were used to review the fire protection program for an operating plant. Whenever applicable, those guidelines were addressed, but to provide broader guidelines for the evaluation of the Robinson plant, the following criteria served as the basis for the overall evaluation:

a) General Design Criterion 3 (10CFR50, Appendix A) - Fire Protection - "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and Control Room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation do not significantly impair the safety capability of these structures, systems, and components."

b) Defense in Depth Criterion - For each fire hazard, a suitable combination of fire prevention, fire detection and suppression capability, and ability to withstand safely the effects of a fire shall be provided. Both equipment and procedural aspects of each shall be considered.

c) General Design Criterion 19 (10CFR50, Appendix A) - Control Room - "A Control Room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCA).

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

d) Single Failure Criterion - A single failure criterion was applied to fire suppression systems which protect systems and equipment important to safety, including equipment required for safe shutdown. The single failure criterion is that no single failure shall result in complete loss of protection of both the primary and backup fire suppression capability.

e) Fire Suppression Systems Capacity and Capability - Fire suppression capability shall be provided, with capacity adequate to extinguish any fire which may credibly occur and have adverse effects on equipment and components important to safe shutdown.

f) Backup Fire Suppression Capability - For those areas which have an identified fire hazard possibly affecting systems or components important to achieving safe shutdown, total reliance for fire protection shall not be placed on a single automatic fire suppression system. Appropriate backup fire suppression capability shall be provided.

g) Occurrence of Fire and Other Phenomena - Fire shall not be considered to occur simultaneously with other accidents, events or phenomena such as a design basis accident. Capability shall be provided (consistent with General Design Criterion 19) to safely shutdown the plant in the event of any single fire which may credibly occur.

9.5.1.1.1 Quality Assurance

CP&L has a corporate Quality Assurance (QA) program in effect as described in Section 17.2. This QA program is applied to the Fire Protection program to the extent defined below:

a) Measures are established to assure that all design-related guidelines of Branch Technical Position 9.5-1 are considered in drafting, design, and procurement documents and deviations therefrom are controlled.

b) The program verifies that inspections, tests, administrative controls, fire drills and training which constitute the Fire Protection program are prescribed by written instructions, procedures or drawings, as appropriate, and are accomplished in accordance with these documents.

c) The program verifies that purchased material, equipment and services conform to the procurement documents. Nonconforming items are controlled to prevent inadvertent use or installation.

d) Records are maintained to provide existence of the performance of all tests, drills, and audits.

Applied as such to the Fire Protection program, the QA program complies with the stated guidelines.

9.5.1.1.2 Effect of Rupture or Inadvertent Operation of the Fire Water System Piping

9.5.1.1.2.1 Containment Building

The fire water piping inside the containment is seismically designed except for the deluge headers in pump bays B and C. However, since these headers are not normally charged, a failure of these headers during a seismic event will not cause flooding. Analyses also show that falling header parts will not cause damage to safety-related components.

The containment vessel isolation valves for fire water are left normally open and are closed automatically on a Phase "A" isolation signal. Due to the seismic design and missile protection afforded the fire water piping, a rupture will not occur from a design basis accident. Inadvertent operation of the deluge system would not present a flooding or impingement hazard since all sprinkler systems are closed systems.

9.5.1.1.2.2 Reactor Auxiliary Building

The fire water system piping has been analyzed to determine the effect of a pipe rupture on safety-related equipment. A detailed analysis for inadvertent actuation was not considered necessary for two reasons. First, a double failure (actuation signal from two independent detector trains, in addition to the spurious or accidental opening of one or more sprinkler heads) in the actuation system would be required for an inadvertent actuation to occur; and second, the only safety-related equipment affected would be the service water booster pumps, which are not part of the safe shutdown system. In addition, spurious actuation of the deluge valve would be alarmed in the Control Room via the flow alarm and total water flow could be limited by operator action. The following description summarizes the effects of various postulated fire water system pipe ruptures on safety-related equipment.

The pipe rupture analysis was performed by considering three separate pipe break scenarios in the Auxiliary Building. These scenarios consisted of a 4 in. pipe break in the hallway on Elevation 246 ft, a 4 in. pipe break in the pipe tunnel on Elevation 226 ft, and a 4 in. pipe break in the hallway near motor control center (MCC) No. 5 on Elevation 226 ft. The postulated break locations were selected to typify the areas with water filled pipe in the Auxiliary Building. A detailed description of the analysis is contained in Reference 9.5.1-1.

The scenario descriptions and analyses show that the floor drain system is essential for protection of electrical safety-related equipment on the second floor. The 4 in. break in the hallway on Elevation 226 is the more severe accident, since it can damage safety-related equipment by direct water impingement. All other breaks could cause equipment damage by flooding and allow time for corrective action to be taken by operators.

9.5.1.1.2.3 Summary

The results of the analyses of the effects of an inadvertant actuation of, or pipe rupture in, the plant fire water system in the Containment Building indicate that no damage to safety-related equipment would occur. The results of the analysis of the Auxiliary Building indicate that damage to safety-related equipment may occur. Inadvertent actuation of the fire system does not pose a threat to safety-related equipment since a double failure is required for actuation and the low flow rate through any open spray heads would not cause building flooding. Pipe ruptures can cause damage to safety-related equipment by both direct impingement and building flooding. All equipment damage that would be caused by building flooding can be eliminated or minimized by prompt operator action to terminate the water flow into the Auxiliary Building by closing the appropriate post indicator valve. CP&L constructed a spray shield to protect MCC No. 5 from immediate damage by direct water impingement from a pipe rupture since operator action could not mitigate the consequences of this rupture. The floor drain system is very important in preventing safety-related equipment damage for a pipe rupture on Elevation 246 but acts as a water distribution system on Elevation 226 due to low (20 gpm) sump pump flow rate and the small sump.

A further evaluation of the consequences of inadvertent operation of the fire extinguishing system is addressed in the description of each respective system and in the detailed fire hazards analysis for each fire area.

9.5.1.1.3 Unusually Hazardous Materials

The amount of combustible materials in the safety-related areas of the plant has been minimized insofar as reasonably possible. Major combustibles in the safety-related areas are lubricants required for rotating equipment and insulation and jackets on electrical cables. The majority of electrical cables use polyvinyl chloride (PVC), except in the containment, where silicone rubber (which has superior fire-retardant properties) is used extensively. Cable trays which contain engineered safeguards cables were coated with a flame retardant mastic to minimize their combustibility. Other combustibles include a day tank containing fuel for each diesel generator; and paper, plastic and cloth (which are generally found in small quantities, if present at all). Noncombustible materials are used for thermal insulation. Compliance with specific guidelines related to control of combustibles is noted below.

a) Isolation and separation of combustibles are discussed under Section 9.5.1.2.1.1a as related to fuel oil and lube oil systems, cable insulation, etc.

b) Bulk gas storage inside structures housing safety-related equipment is not present nor permitted. Small quantities of compressed gases in high pressure bottles [<300 ft^3 at standard temperature and pressure (STP)] are maintained on systems requiring their use inside these areas. These are as follows:

1)	Isolation Valve Seal Water (IVSW) System	2 - 275 ft^3 Nitrogen bottles
2)	Gas Analyzer System	1 - 175 ft^3 Hydrogen bottle 1 - 275 ft^3 Nitrogen bottle 1 - 200 ft^3 Calibration gas bottle
3)	Diesel CO_2 System	19 - 75 lb, 2 - 50 lb CO_2 bottles
4)	Post Accident H_2 Venting Systems	2 - 275 ft^3 Nitrogen bottles
5)	North and South Cable Vault CO_2 System	36 - 75 lb, CO_2 bottles

Low pressure gas storage in the form of air receivers is positioned in the diesel generator rooms and in the lower hallway of the Auxiliary Building. The starting air receivers for the diesels are pressurized to 275 psig and protected by a safety valve. The loss of one receiver will not affect the redundant diesel since each unit and its receiver are located in separate rooms. Two air receivers for station and instrument air are located in the hallway and pressurized to 100 psig. Both are protected by safety valves. Bulk storage of nitrogen in cryogenic form and high pressure cylinders is maintained outside the Auxiliary Building in a separate, open structure. The gas is piped into the Auxiliary Building in one inch carbon steel lines (maximum pressure 600 psig) to the safety injection (SI) accumulators. Bulk storage of hydrogen is in a building which is more than 200 ft from the Auxiliary Building. Storage is by means of a high pressure tube trailer parked parallel to the buildings.

Hydrogen is piped at 120 psig through double wall pipe to the generator in the Turbine Building and the volume control tank in the Auxiliary Building. The double wall pipe is used underground only. Piping running through the building is single wall one inch, schedule 40 pipe.

c) Use of plastic materials has been minimized, except for PVC as cable insulation. Flame retardant coatings are applied to engineered safeguards cables, and detection and extinguishing capability are provided in areas of maximum cable concentration.

d) The only storage of flammable liquids near safety-related equipment is fuel oil day tanks for the diesel generators. Storage tanks are present for diesel oil, auxiliary boiler and turbine oil. The diesel oil storage tank is approximately 100 ft north of the Auxiliary Building and is surrounded by a dike. The fuel tank for the auxiliary boiler is located about 400 ft northeast of the Auxiliary Building. Turbine oil storage is in a nonsafety-related area and is greater than 50 ft away from the nearest shutdown-related equipment, the steam-driven auxiliary feedwater pump.

Bulk storage of flammable liquids is maintained outside of buildings housing safety-related equipment. The oil storage building is separate from all other structures and is located greater than 20 ft from Unit 2. Storage of small amounts of flammable liquids and hazardous chemicals are stored in specially constructed flammable liquid storage cabinets. These cabinets meet the requirements of National Fire Protection Association (NFPA) 30, Section 42, 1976 Edition.

Certain hazardous materials require special protection. These include:

a) Welding and Cutting, Acetylene-Oxygen Fuel Gas Systems

Welding and cutting equipment is stored in shop storage areas betwen Units 1 and 2. These areas are remote from safety-related systems and equipment of Unit 2 and were not considered in the fire hazards analysis. A permit system for use of such equipment is in effect.

b) Storage Areas for Dry Ion Exchange Resin

A dedicated storage area for dry resin is provided, removed from safety-related areas with appropriate protective features.

c) Hazardous Chemicals

Only minimal amounts of hazardous chemicals are present at the plant site and these are generally in small containers in the hot chemistry laboratory storage room. This is a separate fire area containing no safety-related equipment. A flammable liquids storage cabinet is available in this area for particularly hazardous items. Portable extinguishers are available in this area and an automatic fire detection system has been provided for this area.

d) Materials Containing Radioactivity

Spent resin storage is provided by a metal tank in an area free of combustibles. Solid materials containing radioactive material are processed in the solid waste handling room with minimal temporary storage. Materials are ultimately drummed and removed from the site. The protective features of the waste handling room and processes for handling other radioactive materials provide compliance with the guidelines of BTP 9.5-1.

9.5.1.2 Systems Description

9.5.1.2.1 Features of Building Arrangement and Structural Design That Contribute to Fire Prevention and Control

9.5.1.2.1.1 Building Design Features

Buildings and structures containing components and systems important to safety are of reinforced concrete construction. This type construction has the advantages of noncombustibility of construction materials, and also of a high degree of resistance to the effects of fires which might occur. Specific features regarding building design are noted below.

a) The plant layout subdivides the plant into numerous fire areas which contain safety-related equipment. The barriers (walls, etc.) which separate these areas are capable of isolating the areas from fire hazards in other areas. The use of combustibles in safety-related areas has been minimized. They consist primarily of lube oil required for certain equipment, diesel generator fuel and PVC insulation and jackets on electrical cables. Where redundant items of safety-related equipment or cables are not separated, an appropriate combination of fire retardant coatings for cables, fire detection and alarm, and automatic and/or manual fire suppression is provided. In addition, instrumentation, controls, power supplies, etc., are provided so that the plant can be shut down and maintained in a safe condition in the event of fire in any area.

b) Safety-related systems and equipment, and equipment required for safe shutdown have been identified (Section 9.5.1.3.3b). Fire hazards have been identified and a fire hazards analysis made.

c) The cable spreading room is separated from other areas of the plant and is not shared with another plant.

Additional information relative to the cable spreading room is given in Section 9.5.1.2.4.9b.

d) Interior wall and structural components are steel, reinforced concrete, and other noncombustible materials. Thermal insulation and radiation shielding materials are also noncombustible. Interior surfaces in most areas are bare concrete; paint is applied in some areas.

e) Metal deck roof construction has not been utilized.

f) Suspended ceilings are noncombustible and exist only in the Control Room and hot chemistry laboratory and adjacent office area. Only cable in conduit for lighting is present in the Control Room. Combustibles in concealed spaces in the office lab area consist only of insulation for small quantities of electrical cables in trays. Neither the area nor the cables is safety-related.

g) There are no oil filled transformers in safety-related areas.

h) Floor drains are provided throughout the plant as are curbs and pedestals for equipment. Floor and equipment drains are provided with appropriate traps, and holdup and monitoring capability for controlled areas of the plant. Since areas subject to fire hazards are located at or above grade, water will not accumulate and create unacceptable consequences.

i) The walls which form barriers to define separate fire areas are of heavy reinforced concrete or pyrocrete construction. Their design is based on other factors in addition to fire resistance, and meets or exceeds the requirement set by the area fire hazard. Door frames and hardware similarly meet or exceed the requirements of the area hazards. Doors are normally closed and ventilation openings are protected by fire dampers rated to meet or exceed requirements set by the area hazards.

9.5.1.2.1.2 Ventilation System Design Features

Isolation and containment features for flame, heat smoke, hot gases and other contaminants are also important to plant fire protection and control. An effective plant ventilation system is a major factor in this regard. Specific features related to ventilation are noted below.

a) As indicated above, ventilation related to handling the products of combustion were evaluated on the type of fire suppression system and potential for radioactive releases.

b) No ventilation systems specifically designed to exhaust smoke or corrosive gases exist. Such systems are not compatible with gas suppression systems and have potential of violating controlled areas. The use of ventilation system fire dampers will isolate areas and failure or inadvertent operation of dampers would not compromise controlled areas.

c) All ventilation system actuators are remotely controlled from the Control Room by the operator except for fire dampers and where automatic interlocks are involved. All actuators are designed to fail to the position required for post-accident operation upon loss of electric or pneumatic power. Instrumentation in the Control Room will provide information to allow proper remote operation of the system. The Control Room and Cable Room/Relay Room systems and power supplies are separated from the areas they serve.

d) The plant has a total of eight charcoal filters, none of which have sprinkler systems. The use of an automatic sprinkler system has a very negative possibility of inadvertent operation which would completely disable the charcoal filter's function by saturating the bank. Considering the relatively slow propagation rate of a charcoal fire, an automatic system appears unnecessary. Since all ventilation systems with charcoal filters are not in normal use, the fire hazard and consequence of a fire affecting them is

small. When in use, the deposited heat load from radioactive particles is not sufficient to cause combustion. All charcoal filters are housed in metal enclosures and preceded by absolute filters. Since a postulated fire in the charcoal filters would not affect safe shutdown and the filters function of removing radioactive material would be degraded even if sprinklers were provided, use of sprinklers is not necessary.

e) Safety-related areas with separate fresh air intakes from the rest of the plant include the Control Room, Relay Room, Battery Room, and Cable Spread Rooms. These intakes are remote from the exhaust outlets serving other fire areas. Separation between a small exhaust air outlet from the Battery Room to the intake for the Cable Spread Room system is about 25 ft. Most safety-related areas are serviced by the Auxiliary Building system with essentially no influence between supply intake and the exhaust stack.

f) Plant stairways are generally exterior or not enclosed and are not subject to being infiltration paths for smoke. One enclosed stairwell exists for the Auxiliary Building which leads from the ground level to the control level. Doorways at landings provide a three-hour fire barrier to building areas. This stairway is not entirely interior to the building but enclosed on the side of the building and has a door to the outside at the control (upper) level which could be used for venting smoke.

The elevator for the Auxiliary Building is exterior to fire areas with door openings at each level to the outside rather than the building. The Containment Building elevator does not have doors of adequate rating but the escape routes are pre-established using stairways.

g) Smoke and heat vents do not exist in the plant areas and are not proposed due to incompatibility with the need for control of radioactivity and with proposed fire suppression systems.

h) Scott-Air Packs are available for use by the fire brigade and Control Room. These have a service life of 30 minutes maximum. Extra supply bottles are available. An installed compressor and cascade system are available for refilling.

i) Total flooding CO_2 systems are installed for fire suppression in the Diesel Generator Rooms and the North and South Cable Vaults. For these gas suppression systems, damper closure upon initiation of gas flow is automatic.

9.5.1.2.1.3 Identification of Fire Areas

At HBR 2, areas containing components, systems and structures important to safety have been identified. Figure 9.5.1-1 is a plan of the plant showing those areas, which include the following:

a) Reactor Containment Building

b) Reactor Auxiliary Building

c) A small area in the Turbine Building

d) Storage Tanks for condensate, refueling water, diesel oil, clean water, and

e) Intake Structure.

These have been further subdivided into fire areas on the basis of existing walls which adequately serve as fire barriers. Figures 9.5.1-2 through 9.5.1-4 show the plan of the Auxiliary Building, as it has been divided into fire areas. Figures 9.5.1-5 and 9.5.1-6 show the Containment Building fire zones.

Plant fire areas have been established such that the area is either enclosed or bounded by adequate space that is free of combustibles to prevent the spread of fire to other areas. Most areas inside the Auxiliary Building have enclosing floors, walls, and ceilings of sufficient thickness to provide a fire resistance rating of three hours or more.

Where barriers of less than three hour rating were determined to be acceptable, their acceptability was based on other protective features including location, combustible loading and/or consequence of a fire on plant operation and shutdown capability.

Other fire barriers important to effective fire protection and control include fire doors, fire stops for cable penetrations, and fire retardant coatings for electrical cables.

Doors that have been installed in fire barriers have fire ratings commensurate with the requirements of the fire area involved. Daily inspections are made by fire protection technical aides to verify that all fire doors are closed.

Fire stops for cable penetrations are constructed of a mineral fiber board cut to shape around the cable tray. High temperature mineral fiber is packed between the opening in the mineral fiber board and the cables. The fiber and exposed surfaces of the mineral fiber board are coated with a flame retardant coating applied by spraying each side of the penetration.

A sketch of typical construction is provided as Figure 9.5.1-7.

Conduit penetrations contain 9 to 12 in. of an approved silicone foam. Test data provided by the manufacturer, Brand Industrial Corporation, indicate that 9 in. of foam provides sufficient protection to meet the requirement that the conduit penetrations 3 in. or less in diameter have a three hour rating. Larger conduits employ 12 in. thickness of silicone foam.

In addition, cables in critical plant areas outside containment were coated with a flame retardant coating. The flame retardant coating was applied in accordance with the manufacturer's recommendations. During application, any configuration not covered by the manufacturer's standard application recommendations was documented and the manufacturer was consulted to determine an alternate application method for each specific case.

The most common cable and cable tray configurations would be covered by the manufacturer's standard application recommendations. The manufacturer's specific recommendations on less common configurations would be expected to generally provide a comparable level of protection.

Cables inside containment are insulated with silicone rubber which has fire resistance characteristics that are superior to the polyvinyl chloride insulated cable used outside containment.

Plant design and construction features that specifically relate to fire safety are discussed on an area-by-area basis in Section 9.5.1.3.4.

Effective means of access and egress are also needed for effective fire protection and control.

All safety-related areas except the containment are reasonably accessible for manual fire fighting. Escape and access routes, clearly marked, are provided by stairways internal and external to the Auxiliary and Turbine Buildings.

During normal operation, the containment is sealed and access is provided through an interlocked double-door personnel air lock. Special procedures must be followed to gain access, increasing the response time of the fire brigade.

9.5.1.2.2 Requirements for Design of Fire Water Supply and Distribution Systems

9.5.1.2.2.1 Fire Protection Water Supply

The fire suppression water supply system consists of pumps and an underground loop which supplies water to deluge systems, outside fire hydrants and fire hose racks inside the plant. Figures 9.5.1-8 and 9.5.1-9 show the system. All fire protection piping is designed to meet Class I seismic criteria, except the hose station supply pipe mounted on the Auxiliary Building exterior wall and the deluge headers in the containment pump bays B and C.

The fire loop and header system is normally pressurized by the fire water booster pump, which runs continuously. The main fire water loop is supplied by an electric motor-driven fire pump with backup by a diesel engine-driven pump. Lake Robinson is the water source. Two separate and independent pressure switches allow for automatic operation of the pumps. One switch initiates the electric motor-driven pump and the other initiates the engine-driven engine pump. Both pumps are manually stopped. The system is designed to maintain loop pressure so that, in the event of several fires at the plant, sufficient pressure is maintained at the fire hose stations, fire hydrants, and automatic deluge system.

The fire header is sized to deliver an adequate quantity of water throughout the plant to service all outlets. The yard piping consists of an underground twelve inch supply header to the ten inch diameter fire water line loop header, which completely surrounds all plant buildings. The fire main header supplies water to the following areas:

a) one - 8 in. to main transformers area with a 6 in. lateral to the containment vessel and Auxiliary Building

b) one - 8 in. to the southwest side of the Turbine Building

c) one - 6 in. to the northwest side of the Turbine Building

d) one - 6 in. to the Fuel Handling Building and hot machine shop

e) one - 6 in. to the Reactor Auxiliary Building

f) one - 6 in. to the east side of the Turbine Building

g) five - 6 in. for the fire hydrants

h) one - 4 in. to the intake structure hydrant

Valved branches from the underground fire loop and header system supply interior fire protection systems in the enclosed sections of the plant, including the Turbine Building. Sectionalizing valves in the yard piping system are provided to permit partial pipeline isolation without interruption of service to the entire system during maintenance or future extension of facilities. Fire department connections are provided outside the building through the use of hydrant connections.

Fire protection is provided to the exterior plant areas by six yard fire hydrants. The fire hydrants have two 2 1/2 in. independent outlet connections with gate valves. Each fire hydrant is furnished with 150 ft of 2 1/2 in. flexible single jacket rubber lined hose, one 2 1/2 in. adjustable fog nozzle, a 1 1/2 in. hose, and a 1 1/2 in. gated "y" connection.

Specific guidelines of BTP 9.5-1 relating to the fire protection water supply are given below:

a) The fire main loop piping uses cast-iron pipe underground and aluminum or carbon steel pipe above grade. Loop piping is Class 22 cast iron with mechanical joints. All loop section and unit fire water cross-connection isolation valves have visually indicating posts. The fire main system piping is also separate from service and sanitary water system piping.

b) The Robinson site is shared by Unit 1 (fossil) and Unit 2 (nuclear) with cross-connection to the Unit 1 fire pump available for Unit 2. An eight-inch water line from the Unit 1 fire protection system is connected to the main loop for emergency use, with isolation valves which are normally locked closed. In addition, an offsite fire department could pump water from the discharge canal into the fire loop. A suitable adapter coupling is provided in the nearest hose house for connection between the pumper and the fire hydrant.

c) Upon demand (reduction of header pressure in the fire water system loop), an adequate supply of fire water from the cooling water lake is supplied to the system by an electric motor-driven pump of 2500 gpm capacity at a 290 ft head. This pump is backed up by a separately located internal combustion engine-driven fire water pump of the same capacity, with automatic starting and with alarms in the Control Room. The headflow curve for these pumps is shown in Figure 9.5.1-10. The combustion engine local fuel supply capacity is designed for at least eight hours of operation. Supply from both of these pumps connects to the southeast corner of the loop, but with a separate supply available from the Unit 1 fire pump with connection near the northeast corner of the loop.

d) The source of supply for the fire water system is Lake Robinson rather than tanks. See Item f) below.

e) The flow delivery capability of the fire water system has been independently calculated. The methods and parameters recommended in National Fire Protection Association (NFPA) 15 were used for this calculation. The results indicate that the system is capable of maintaining the pressure in the plant loop at 70 psi or higher with the largest deluge system in operation and with the system supplying an additional 1000 gpm to hoses. The value of the Hazen and Williams friction loss coefficient was taken to be 100, in accordance with NFPA recommendations. In actual service, the loss coefficient is considerably higher (pressure drop lower) for new pipe and would be expected to reach this value only after twenty years or more of service (Reference 9.5.1-2).

f) Lake Robinson provides the source of water for fire protection. The individual Unit 2 fire pumps take their suction from the circulating water intake system on the clean side of the traveling screens between the screens and the circulating water pumps. This configuration provides adequate redundancy for water supply with additional capability available from the Unit 1 fire pump at a separate intake structure. Lake Robinson is also the ultimate heat sink with sufficient capacity to provide fire protection water requirements. Failure of the fire protection system also would not degrade the function of the ultimate heat sink.

g) Hydrants are separated at distances of about 250 ft, except at the north end of the plant, where the distance between hydrants is somewhat larger. All hydrants for Unit 2 are individually connected to the main fire water loop and equipped with two 2 1/2 in. independent connections with gate valves. Five hose houses are located near hydrants and one at the intake structure. A minimum equipment supply is maintained in each house, with more available from stock. Standard fire hose threads are used on all fire protection equipment.

9.5.1.2.2.2 Water Sprinklers and Hose Standpipe Systems

Automatic water deluge systems and hose stations are installed at the plant to provide water suppression capability for areas of significant fire hazard and general fire fighting. Additional pre-action sprinkler systems are installed for the electrical penetration area, reactor coolant pump (RCP) bays A, B, and C, solid waste handling room and portions of the Auxiliary Building ground level hallway. The automatic deluge systems protect the following subsystems:

a) Main Transformers

b) Unit Auxiliary and Startup Transformers

c) Turbine Oil Hazard Area, and

d) Hydrogen Seal Oil Unit and Hydrogen Manifold Area.

The above equipment and areas are protected by means of water deluge valves, spray nozzles and heat actuated devices. These devices when actuated will trip open the pneumatically operated deluge valve in the branch supplying water to the spray nozzles. In addition to the automatic protection, the turbine oil hazard area and the hydrogen seal oil unit and hydrogen manifold areas are each arranged for manual activation by means of a remotely located actuator control station located on the fire alarm box in the SW corner of the Hagan analog room.

The pre-action sprinkler system consists of a deluge valve, fusable link sprinklers, and an actuation system. The pre-action sprinkler system is automatically actuated by redundant signals from two separate trains or may be manually actuated.

Floors in the turbine area are pitched so that any oil spills will drain to proper drainage facilities, thus preventing the spread of an oil fire outside the turbine area. Curbs are also provided to limit the spread of oil.

Two electric-driven air pumps, one for Subsystems A and B, and one for Subsystems C and D, are furnished to maintain supervising air pressure in the thermo-pneumatic system.

Two water-motor fire alarm gongs and two electric trouble alarm horns are furnished for Subsystems A and B, and C and D.

A total of thirty-six hose stations are located throughout the entire plant. The Containment Building has eight stations, the Turbine Building has fifteen stations, with six on the ground floor, five on the mezzanine and four on the turbine level. The Auxiliary Building has ten stations, the Fuel Handling Building has one and the hot machine shop has two hose stations. Each hose station has readily accessible 1 1/2 in. hose lines and continuous flow type hose reels, and these are distributed as stated above so that all areas in the plant are within 20 ft of a spray nozzle. All nozzles are 1 1/2 in. spray nozzles pinned to the hose adapters with provision made to prevent the use of the straight stream position. Electrically safe hose nozzles are installed at hose stations near areas with electrical hazards. In places where automatic suppression systems are installed, manual hose stations can serve as backup protection.

Specific guidelines related to water sprinklers and hose standpipe systems are given below:

a) The deluge systems described above have independent connections to the main fire water loop. Pre-action sprinkler systems and fire hose station standpipes are fed from branch headers that are connected at one end only to the fire water loop. Several stations are connected to each branch header. For water sprinkler systems, piping connections are such that single failure will not impair both primary and backup suppression capability except for fire water in the containment where water enters through a single penetration sleeve and has a single supply line. Each deluge system is equipped with an outside screw and yoke gate valve with water flow alarms as indicated above.

b) Electrical supervision to monitor the position of fire water system control valves is not provided; however, a means of sealing these valves open is provided, and this in combination with administrative controls and periodic inspections assures that valves are maintained open.

c) The automatic water sprinkler system features are consistent with NFPA 13 and 15.

d) As indicated above, interior hose stations are appropriately located and equipped to supply effective hose streams for primary or backup fire suppression.

e) Proper hose nozzles have been supplied at each hose station based on the type of fire hazards in nearby areas. Hose cabinets have markings identifying the applicable use of nozzles and necessary precautions.

f) Fixed foam suppression systems have not been selected for use in any plant areas since other adequate suppression systems are available. Portable foam suppression capability is provided for areas with large flammable fluid concentrations.

9.5.1.2.3 Fire Protection System Design and Installation Codes and Standards

Industrial codes and standards used for design and installation of the plant fire protection system include the following:

a) ANSI - American National Standards Institute

- A13.1 - Scheme for the Identification of Piping Systems
- B2.1 - Standard for Pipe Threads
- B16.9 - Factory Made Wrought Steel Butt Welding Fittings
- B16.11 - Forged Steel Fittings, Socket-welding and Threaded
- B16.25 - Butt Welding Ends
- B16.34 - Steel Valves (Flanged and Butt Welded End)
- B16.5 - Steel Pipe Flanges and Flanged Fittings
- B31.1 - Power Piping
- C1 - National Electrical Code (NFPA 70)
- C33.84 - Safety Stand
- C80.1 - Specification for Rigid Steel Conduit, Zinc Coated
- C80.4 - Specification for Fittings for Rigid Metal Conduit and Electrical Metallic Tubing (UL514)

B512 - Protective Coatings (Paints) for the Nuclear Industry

N101.2 - Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities

N101.4 - Quality Assurance for Protective Coatings Applied to Nuclear Facilities

b) AISC - American Institute of Steel Construction "Specification for the Design Fabrication and Erection of Structural Steel for Buildings," sixth Edition 1963.

c) ASME 77 - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Winter Addendum, 1979.

d) AWS - American Welding Society

D1.1 - 1960 Standard Code for Welding in Building Construction

e) ASTM - American Society for Testing and Materials

A-36 - Structural Steel

A-37 - Malleable Iron Castings

A-53 - Pipe, Steel, Black and Hot Dipped, Zinc Coated Welded and Seamless

A-105 - Forgings, Carbon Steel, for General Purpose Piping

A-106 - Seamless Carbon Steel Pipe for High Temp. Service

A-123 - Specification for Zinc (Hot Galvanized) Coatings on Products Fabricated from Rolled, Pressed, and Forged Steel Shapes, Plates, Bars, and Strip

A-126 - Gray Iron Castings for Valves, Flanges and Pipe Fittings

A-181 - Forgings, Carbon Steel, for General Purpose Piping

A-194 - Carbon and Alloy Steel Nuts for Bolts for High Pressure and High Temperature Services

A-197 - Cupola Malleable Iron

A-234 - Piping Fittings of Wrought Carbon Steel and Alloy Steel for Moderate and Elevated Temperatures

A-307 - Carbon Steel Externally and Internally Threaded Standard Fasteners

A-569 - Specification for Steel, Carbon (0.15 Maximum Percent), Hot Rolled Sheet and Strip, Commercial Quality

E-119 - Fire Tests of Building Construction and Materials

E-376 - Coating Thickness by Magnetic-Field or Eddy-Current (Electromagnetic) Test Methods, Rec. Practice for Measuring

E-386 - Zinc Coating (Hot Dipped) on Assembled Steel Products

f) FM - Factory Mutual Research
- Factory Protection Equipment Approval Guide

g) HIS - Hydraulic Institute Standards

h) IEEE - Institute of Electrical and Electronics Engineers

i) NEMA - National Electrical Manufacturers Association

IS1.1 - Enclosures for Industrial Controls and Systems
ML1 - Metal Framing

j) NFPA - National Fire Protection Association

10 - Portable Fire Extinguishers
12 - Carbon Dioxide Systems
12A - Halon 1301 Systems
13 - Sprinkler System Installation
14 - Standpipe and Hose Systems
24 - Outside Protection
27 - Private Fire Brigades
51B - Cutting and Welding Processes
72D - Proprietary Signaling Systems
72E - Automatic Fire Detectors
198 - Fire Hose, Care of
Fire Protection Handbook

k) NML - Nuclear Mutual Limited

Property Loss Prevention Standards for Nuclear Generating Stations

l) OSHA - Occupational Safety and Health Administration

m) PFI - Pipe Fabrication Institute Standards

ES-3 - Fabricating Tolerances
ES-24 - Pipe Bending Tolerances

n) SSPC - Steel Structures Painting Council

SSPC-SP1 - Solvent Cleaning
SSPC-SP2 - Hand Tool Cleaning
SSPC-SP3 - Power Tool Cleaning
SSPC-SP6 - Commercial Blast Cleaning
SSPC-PA1 - Shop Field and Maintenance Painting

o) UL - Underwriters Laboratories, Inc.
Fire Protection Equipment List

UL1 - Flexible Metal Electrical Conduit
UL360 - Liquid-Tight Flexible Steel Conduit
UL467 - Electrical Grounding and Bonding Equipment
UL6 - Rigid Metal Electrical Conduit

p) South Carolina State and Local Fire Codes

q) NUREG - 75/087
NUREG - 0050

9.5.1.2.4 General Description of Fire Protection Systems

The Fire Protection Systems (FPS) are provided to protect personnel and property in the event of fires. The FPS have the design capacity to detect and extinguish fires which might occur at the station in safety-related areas or which could affect safety-related areas. The fire detection system uses both heat and products of combustion detectors in appropriate locations. A suitable combination of manual and automatic suppression systems is provided. The plant fire suppression systems include the water supply system, deluge spray systems for transformers and turbine oil systems, pre-action sprinkler systems, hose stations, yard hydrants, high pressure carbon dioxide systems, halon system, and extinguishers. See Tables 9.5.1-1 through 9.5.1-3 for the fire protection system documentation and instrumentation data.

9.5.1.2.4.1 Fire Detection Actuation System (FDAS)

The FDAS, illustrated in Figure 9.5.1-11 provides centralized control of the detection, annunciation, and actuation for most H. B. Robinson Fire Protection Systems. FDAS is composed of two independent, and thus redundant, detection and actuation sub-systems, designated train "A" and train "B". Each of these trains consists of its own detectors, alarms, control devices, and annunciator/status indicators.

The FDAS provides the following primary functions:

a) Zone fire detection (via individual zone fire detection devices)

b) Controlled activation of specific zone fire suppression systems, and

c) Zone fire alarm and status indications.

FDAS consists of the following major components:

a) Four Fire Detection and Actuation Panels (FDAP) designated FDAP-Al, A2, Bl, and B2. FDAP locations are:

1) FDAP-Al - Auxiliary Building Hallway near Air Compressor

2) FDAP-Bl - Auxiliary Building Hallway near Component Cooling Room

3) FDAP-A2 - Emergency Switchgear Room

4) FDAP-B2 - Second floor Auxiliary Building in access area around Emergency Switchgear Room

b) Two Fire Alarm/Status Panels (FAP), designated FAP "A" and FAP "B", located in the Control Room

c) Two Reactor Turbine Generator Board (RTGB) annunciator panels, designated BETA-ALARM "A" and "B", located in the Control Room

d) One Containment Fire Protection Status Panel (CFPP) located in the Control Room, and

e) Numerous fire detection devices which are located throughout the plant.

FDAS is designed to provide redundant actuation capabilities. This is accomplished by using design parameters known as one-out-of-two and two-out-of-two coincidence. One-out-of-two coincidence initiated when only one of the two FDAS trains is activated. This condition is used for system tests, status conditions, actuation of Diesel Generator Room dampers and doors, and manual actuation. Two-out-of-two coincidence is initiated when both train "A" and "B" have been activated. This method is used to automatically actuate the fire suppression systems.

The combination of cross-zone detector installation and the actuating parameter "two-out-of-two coincidence" ensures that a suppression system will not be inadvertently activated by the momentary application of a high heat source (e.g., welding equipment, etc.) to a single detector. The single detector sensing an alarm condition does provide "one-out-of two coincidence actuation which will produce the following sequence of events:

a) The local zone audible alarm will sound

b) The affected zone will alarm in the Control Room at the appropriate FAP and Beta-Alarm (indicating a single train has sensed a fire hazard)

NOTE: If zone 24, 25A, 25B, or 25C is the affected area the CFPP will annunciate the affected zone in conjunction with the FAP and Beta-Alarm.

c) The FDAP will sound an internal audible alarm and indicate the affected zone.

FDAS is designed for redundant fire detection by using a system in which the individual detectors of each fire protection train are installed utilizing an installation technique called cross-zoning. This means that the electrical hookup of the detection devices in each zone is alternated between the two fire detection and actuation trains.

The following is a description of the major panels of the FDAS:

a) Fire Detection and Actuation Panels

The Fire Detection and Actuation Panels (FDAP) are divided into two groups for different fire zones as shown by Figure 9.5.1-11.

Each FDAP consists of a variety of modules used to perform alarm, control, status, and annunciation functions.

The control module provides basic control and power indications for FDAS. The status and condition indications provided are:

1) Power - This "Amber" indicating lamp will remain illuminated when normal input power is furnished from either MCC-10 or MCC-9 as appropriate.

2) Alarm - This "Red" indicating lamp will be illuminated when an alarm condition is sensed by a detection device in any fire zone. The FDAP internal audible alarm will sound when this alarm lamp is illuminated.

3) Alarm Silence - This "Red" Light Emitting Device (LED) will be illuminated if the alarm silence has been operated to silence the internal audible alarm.

4) Trouble - This "Amber" lamp will illuminate and the internal audible alarm will sound for any of the following conditions:

(a) Removal of or trouble associated with a system bus connected module

(b) An open in the alarm relay coil

(c) A ground fault on any external line or DC power, and

(d) Improper condition of battery or charger.

5) Ground Fault - This "Amber" LED will illuminate in conjunction with the trouble lamp if the trouble is a ground fault condition.

b) Fire Alarm/Status Panels (FAP)

The Fire Alarm/Status Panels are divided into two sections. FAP "A" is the Control Room annunciator panel for FDAP A1 and A2, while FAP "B" is the Control Room annunciator panel for FDAP B1 and B2. The alarm and status indications provided by FAP "A" and "B" are:

1) System Normal - When illuminated indicates that the associated FDAS train is operating in the designed manner.

2) System Trouble - Illuminated when a trouble develops in a particular train; the appropriate zone identification panel will also be illuminated.

3) System Fire - Illuminated when a zone detector has been actuated; the appropriate zone identification panel will also be illuminated.

4) Manual Actuation - Illuminated when a particular zone fire suppression system has been manually actuated at a FDAP.

5) Actuated - When illuminated, this indicates that the associated zone fire suppression system is actuated.

6) Not Actuated - Illuminated until the associated fire suppresssion system has been actuated.

7) Inhibit - Indicates that a fire zone has been placed in the Inhibit mode at the FDAP to prohibit operation; the appropriate zone identification panel will also be illuminated.

8) Damper Supply Failure - Illuminates when there is a trouble in the damper actuating or relay circuitry.

9) Zone Identification panels - The zone identification panels are labeled by room title on FAP "A" and FAP "B". These status panels illuminate in conjunction with the associated alarm/status indications to identify the affected zone.

10) Local Signal Silence - This "Amber" lamp is illuminated when the Local Signal Silence switch is placed in the "Silence" position. When in the silence position the associated FAP audible alarm is silenced.

c) Beta-Alarm

The Beta-Alarm is located at the Control Room RTGB. This alarm panel is used to initially alert Control Room operators of a fire hazard, system trouble, or abnormal condition. The Beta-Alarm is divided into two sections; Beta-Alarm "A" and "B" for the respective FDAS trains. The alarm and status indications provided by the Beta-Alarms are:

1) Normal - The associated FDAS train is operating in its designed manner

2) Abnormal - A problem exists in FDAS train "A" or "B". The appropriate FAP will indicate the specific zone and the particular trouble area

3) Inhibit - A fire zone has been placed in the inhibit mode at the FDAP. The appropriate FAP will indicate the specific zone

4) Fire - At least one FDAS train has sensed a fire hazard and is in the alarm condition

5) Loss of Air Pressure in Auxiliary Building Hallway - The supervisory header for this water deluge system has been depressurized

6) Water Flow Auxiliary Building Hallway - The water deluge suppression system for this area has been actuated

7) Loss of Air Pressure Solid Waste Room - The supervisory header for this water deluge system has been depressurized

8) Water Flow Solid Waste Room - The water deluge suppression system for this area has been actuated, and

9) Hagan Room Supervised Air - Low air pressure or Hagan Room dry standpipe value.

NOTE: All other Alarm/Status indications on the Beta-Alarm panels do not concern Fire Protection Systems.

d) Containment Fire Protection System Panel (CFPP)

The CFPP provides control functions and alarm status, indication for those fire zones located in the Containment Building i.e., zones 24, 25B, and 25C.

The control functions provided by the CFPP are:

1) RCP and Penetration Isolation Valve - Spring - return switch for isolation valves FP-248, 249, 256, and 258. The normal isolation valve position is "Open" and is idicated by the "Red" lamp being illuminated. When the manual switch is operated to the "Close" position, the respective isolation valves will be closed and the "Green" lamp will be illuminated. When in the closed position no water can be supplied to the Containment Pre-Action Water Deluge System or containment vessel hose stations.

2) Penetration, RCP "A", RCP "B", and RCP "C" Deluge Valves - Spring - return switch for the respective fire zone deluge valves FP-324, 349, 374, and 399. When these switches are operated, the associated deluge valves is opened, the deluge flow alarm will be initiated, and water will be present at that system's pre-action sprinkler heads. This is the Manual Actuation Mode for Containment Building fire zones.

The CFPP Alarm and Status indicators illuminate to indicate:

1) Deluge Pressure Lo - The deluge header for the respective fire suppression system has been depressurized

2) Deluge Flow - The deluge valve is open and water is flowing in the indicated deluge system

3) Fire - The indicated containment detection train has been activated

4) Inhibit - The indicated fire zone has been placed in the Inhibit mode at the FDAP

5) Fire Protection System Air Pressure Lo - The supervisory air source for the Containment Building Pre-Action Water Deluge Systems is low, and

6) Flasher Module Indicator Power On - Normal power is supplied to the CFPP. When the "Test" pushbutton is depressed this status indicator will flash and the audible alarm will sound.

9.5.1.2.4.2 High Voltage Fire Detection System

The High Voltage Fire Detection System is designed to detect either visible or invisible smoke (or other products of combustion) and/or heat. When products of combustion or heat are detected, a local alarm is sounded and indicating lamps will light in both the zone indicating units in the Emergency Switchgear Room and in the fire alarm box in the southwest corner of the Hagan Room. An amber "TROUBLE" lamp light on the fire indicating unit and a bell will ring in the Emergency Switchgear Room in the event of a failure which makes the system inoperative.

The system utilizes ionization detectors, thermal detectors, two zone indicating units, two fire alarm bells, a trouble bell, a fire indicating unit and fire alarm box.

Ionization detectors (smoke detectors) and thermal (heat) detectors are installed in the areas listed below.

Area	No. of Heat Detectors	No. of Smoke Detectors
Electrical Equipment Area	4	2
Battery Room	1	1
Safeguards Relay Rack Area	2	2
Cable Spread Room	2	2
Penetration Cable Vault South	2	2
Penetration Cable Vault North	1	1
Rod Control Room	3	0

The following is a description of the major components used in the High Voltage Fire Detection System.

a) Fire Indicating Unit

The Fire Indicating Unit is mounted on the south wall of the Emergency Switchgear Room directly above the zone indicating units. This unit is the basic power control unit for the fire detection system and provides the following functions:

1) Furnishes necessary DC power for operation of detectors and zone units

2) Indicate fire alarm signals

3) Indicate trouble signals, and

4) Provides power for the detector visual indicators.

Three indicating lamps are provided on this unit for power, fire, and trouble. The unit contains an "on-off" switch for applying power and resetting the system. Separate switches for silencing audible fire alarm and trouble signals are provided.

b) Ionization Smoke Detectors

The Ionization Smoke Detectors used in this system consist of two ionization chambers and the necessary related amplification circuits. The ionization detector has, as a sensing element, an ionization chamber which utilizes the principle that air can be made electrically conductive (ionized) by bombardment of the nitrogen and oxygen molecules with alpha particles, emitted by a minute source of radioactive material. A voltage applied across the ionization chamber causes a very small electrical current to flow as the ions travel to the electrode of opposite polarity. Combustion particles enter the chamber and attach themselves to the ions, causing a reduction in the number of ions and thus a reduction in current flow. The reduced current flow increases the voltage which upon reaching a predetermined level, activates the alarm device.

c) Thermal Heat Detectors

The al Heat Detectors are a combined rate-of-rise and fixed temperature type. 1..ese detectors operate in the following manner: Heat on the outside of the chamber causes the air in the chamber to expand. When the air expansion rate exceeds the capacity of the vent to relieve the pressure, the diaphragm is flexed to close the electrical contacts it contains. Slow changes in pressure, caused by ambient temperature changes near the chamber, are vented to atmosphere, and the diaphragm is not expanded sufficiently to cause an alarm. The device also contains a fusible alloy which will melt at a predetermined point and allow the electrical contacts to close.

d) Zone Indicating Units, Alarms, Fire Alarm Panel

Two Zone Indicating Units are mounted on the south wall of the Emergency Switchgear Room. These units subdivide the alarm or trouble signals by zones so that the fire or trouble may be readily located. In addition, the indicating units have been modified to include alarm test push buttons which are used to manually test the unsupervised (i.e., no electrical monitoring system) cables from the Zone Indicating Units to the Fire Alarm Panel in the Hagan Room.

Two fire alarm bells are installed in the Reactor Auxiliary Building, one in the Cable Vault and one in the Cable Spreading Room which are actuated upon detection of a fire. A trouble bell is mounted on the south wall of the Emergency Switchgear Room, and will sound for open or short circuits, loss of power, under voltage and ground faults. A Fire Alarm Panel is mounted in the southwest corner of the Hagan Room. Lamps are mounted on the panel to indicate the following:

1) Smoke Detector trouble
2) Electrical Equipment Area Fire
3) Transformer Yard - H_2 Area Fire
4) Turbine Oil Area Fire
5) Generator A Fire
6) Generator B Fire
7) Rod Control Room Fire
8) Cable Vault North Fire
9) Cable Vault South Fire
10) Cable Spreading Room Fire
11) Safeguard Relay Rack Area Fire

The appropriate indicating lamps will illuminate in the event a fire is indicated in any of the electrical equipment rooms or in the event that the automatic sprinkler systems are actuated. In addition, control switches are provided for the deluge systems in the Turbine Oil Hazard Area and the H_2 Seal Oil Unit Area and for the fire water pump. Two local water motor fire alarm gongs and two electric trouble alarm horns are provided for the deluge systems. A buzzer will sound in the Hagan room and the RTGB will annunciate if any of the indicating lights on the fire alarm panel are energized.

9.5.1.2.4.3 Fire Doors

The Fire Doors used are two types, both designed to provide fire protection for a period of 3 hr. The majority of the doors are the Flush Mounted Swinging Door type which are equipped with automatic door return (closing) mechanisms. The second type of fire doors used are the Horizontal Sliding Fire Door.

An electrical Fire Door Supervision System will annunciate and alarm at the Fire Door Supervision System Control Panel in the Control Room when a supervised fire door is held open for a period of approximately 5 min. This is accomplished by using a link delay and a door limit switch which is installed in a manner such that a normally closed contact in the switch is opened when the door is opened.

SUPERVISED FIRE DOORS

Fire Door	Location
FD-6	East door to South Cable Vault (motor generator (MG) Set Room)
FD-7	West door to South Cable Vault (MG Set Room)
FD-8A	East door to North Cable Vault
FD-8B	West door to North Cable Vault
FD-11	Battery Room to Emergency Switchgear Room
FD-13	Stairwell to 2nd floor Auxiliary Building
FD-14	Controlled Access around Emergency Switchgear to 2nd floor Auxiliary Building Hallway
FD-15 A & B	Emergency Switchgear Room to 2nd floor Auxiliary Hallway
FD-16	Rod Control Room
FD-21	Unit 2 Cable Spreading Room to Safeguards Room
FD-44	Mezzanine Deck to Cable Spreading Room
FD-45	Mezzanine Deck to Emergency Switchgear Room

9.5.1.2.4.4 Fire Dampers

Fire dampers are an integral part of the overall fire protection program. These dampers are installed in the heating, ventilation and air conditioning (HVAC) ducts, wherever those ducts penetrate a fire barrier. The purpose of the fire dampers is to:

a) Prevent the spread of smoke and other combustion by-products from an initial location to other areas via the HVAC ducts

b) Prevent the spread of fire by acting as a physical barrier equivalent to the fire protection capability of the structure which is penetrated by the duct

c) Seal an affected room so that the extinguishing concentration of an installed fire suppression system (i.e., Halon, CO_2) is not diluted by leakage via the HVAC ducting, and

d) Seal an affected area to prevent the HVAC ducting from providing an air (oxygen) replenishment source to a fire.

The fire dampers utilized are of the fusible-link type and the automatic closure type.

The fusible-link actuated fire dampers can be either a vertical or horizontal installation. In either case the fire damper blade is held in the open position by a fusible-link mechanism. This fusible-link is designed to melt, and thus release the blade, when the surrounding ambient duct temperature reaches 165°F.

Automatic fire damper installations are used for those HVAC ducts penetrating areas which have an installed gas fire suppression system. The purpose of this type damper is to provide a means for shutting off the ventilation flowpath of an area prior to the initiation of that area's fire suppression system. To accomplish this, the automatic type fire dampers use both an Electro-Thermal Link and a frangible-link. To provide operational redundancy these two actuation devices are connected in series such that either one can release the fire damper blade.

The Electro-Thermal Link is a fusible-link type device which can be electrically heated to its 165°F melting point. This heating is normally accomplished by an electrical input from the FDAS; however, the Electro-Thermal Link will also melt if the surrounding ambient temperature reaches 165°F. The frangible-link releasing device uses an electrically actuated, small explosive charge. When an electrical actuating signal is received from FDAS, the explosive charge detonates and releases the fire damper blade.

9.5.1.2.4.5 Low Voltage Fire Detectors

The purpose of fire detectors is to detect either fire and/or combustion by-products and provide an output signal to other fire protection systems; e.g., FDAS. There are numerous fire detectors installed. More thau one type of detection device is used in all zones to provide for more reliable fire detection. The fire/combustion by-product detectors used are:

a) Low Voltage Ionization Detectors

b) Photo-Electric Smoke Detectors

c) Ultraviolet Flame Detectors

d) Infrared Flame Detectors, and

e) Thermal Fire Detectors.

The type of detection device is described below:

a) Low Voltage Ionization Detectors

The Low Voltage Ionization Detectors are designed to detect nonvisible combustion by-products so that heat or flame is not required prior to activation of the alarm sequence. These detectors are installed in all plant fire zones except 1, 2, 14, 18, 21, and 26.

b) Photo-Electric Smoke Detectors

Photo-Electric Smoke Detectors respond to visible smoke concentrations. This type of detector is used in the following locations:

1) Zone 9 - North Cable Vault

2) Zone 10 - South Cable Vault

3) Zone 12 - Auxiliary Building Hallway

4) Zone 19 - Cable Spreading Room

5) Zone 20 - Emergency Switchgear Room

6) Zone 26 - Containment Air Recirculation

7) Zone 27 - Residual Heat Removal Room

A Photo-Electric Smoke Detector is constructed of a Light Emitting Device (LED) and a photo-electric cell arranged in such a manner (inside a labyrinth chamber) that light from the LED can strike the photo-electric cell only when it is scattered by visible smoke particles. The detector is designed to delay activation until the smoke has been present for approximately 5 to 10 sec, to avoid inadvertent activation by an ambient light source combined with a temporary smoke concentration.

c) Ultraviolet Flame Detectors

This detector is designed to be relatively insensitive to solar radiation and normally cannot be activated by artificial light. The detector is essentially a geiger-mueller (GM) gas type cathode tube that is designed to detect flame radiation energy in the ultraviolet spectrum, below the range of human vision. This unit will detect flame radiation energy at a wave length between 1850 and 2450 Angstroms. These detectors are only used in Diesel Generator Rooms "A" and "B".

d) Infrared Flame Detector

An infrared flame detector responds to flame radiated energy in the same basic manner as the ultraviolet detector described previously: the primary difference being the wave-length of the flame radiated energy it is designed to receive. This detector receives energy with wave-lengths above 7,700 Angstroms (which is above the range of human vision).

The detector is constructed of a lead sulfide photo-electric cell located behind an infrared filter lens. In response to flame radiated energy, the photo-electric cell generates an electric current that is amplified and used to activate the alarm sequence. The detectors are used only in the reactor coolant pump bays.

e) Thermal Fire Detectors

The Thermal Fire Detector is a rate compensated/fixed temperature (135^{o}F) device. This type of detector is the one used most extensively at HBR.

Thermal Fire Detectors are installed in the following areas:

a) Solid Waste Handling Room

b) Heating and Ventilation Equipment Room for Control Room

c) Auxiliary Building Hallway, Second Level

d) Safety Injection Pump Room

e) Charging Pump Room

f) Component Cooling Pump Room

g) Hot Chem Lab

h) Auxiliary Feedwater Pump Room

i) Boron Injection Tank Room

j) Auxiliary Building Hallway, Ground Level

  1) Near Diesel Generators
  2) Near Air Compressors
  3) Near Component Cooling Pumps

k) Control Room

l) Relay Room

m) Diesel Generator Room A*

n) Diesel Generator Room B*

o) Residual Heat Removal (RHR) Pit (outside Auxiliary Building)

p) Containment, Cable Penetration Area

q) Primary Coolant Pumps

r) Containment, Air Recirculation Units

s) North Cable Vault

t) South Cable Vault

u) Cable Spread Room

v) Emergency Switchgear Room

w) Battery Room, and

x) Pipe Alley.

*These detectors supplement detectors associated with the CO_2 system.

9.5.1.2.4.6 Halon Fire Suppression System

9.5.1.2.4.6.1 Description

The Halon 1301 fire extinguishing system provides a permanently installed automatic means of fire extinguishment for the Unit 2 Cable Spreading Room and the Emergency Switchgear Room.

The Halon 1301 extinguishing system is designed to supply a 5 percent atmospheric concentration of halon to the selected hazard zone for fire extinguishment. To achieve this concentration, the Cable Spreading Room requires 258 lb of halon (four of the halon cylinders) and the Emergency Switchgear Room requires 658 lb of halon (ten of the halon cylinders). This design concentration can only be assured by closure of the associated zone ventilation dampers and doors. Normal leakage around closed doors is expected and does not prevent the system from obtaining the design concentration.

The cylinders are divided into two distinct banks of 10 each, i.e., a main bank of cylinders numbered A1 through A10 and a reserve bank of cylinders numbered B1 through B10. Either bank is sufficient to provide the required halon concentration; therefore, the reserve bank is a redundant halon supply. Main-reserve switches are provided to select the appropriate cylinders for both Rack 1 and Rack 2 to maintain continuous fire protection if a trouble exists in any portion of a cylinder bank. The cylinders are further divided into two separate functional banks. Cylinders A1 through A5 are utilized to provide the fast flooding function, while cylinders A6 through A10 provide the extended flow mode of operation. The reserve bank (B1 through B10) is functionally identical.

9.5.1.2.4.6.2 Operation

The Halon Fire Suppression System is normally in a ready standby status which provides for discharge of halon upon receipt of a "two-out-of-two coincidence" signal from the FDAS. In this state, the FDAS is clear of all abnormal conditions. Normal (automatic) operation is accomplishd by the following sequence:

a) A zone has at least one fire detector from both train "A" and "B" to sense a fire hazard condition.

b) When both the "A" and "B" trains of the FDAS have sensed the hazard condition, the time delay module in the FDAP starts a 25 sec time delay. Simultaneously, a local fire alarm is sounded in the affected fire zone and the hazard is annunciated at the FAP and the Beta-Alarm in the Control Room. Upon completion of the 25 sec time delay, an actuation signal is supplied to the appropriate solenoid actuator on the fast flooding cylinder by FDAP.

c) The fast flooding subsystem will completely discharge in approximately 10 sec. As the fast flooding cylinders are actuated, the extended flow cylinders are actuated and maintain the 5 percent halon concentration for the required 30 min soak period. Once actuated, the halon continues to discharge until the cylinders are exhausted.

The system may also be manually operated at the FDAP or it can be manually actuated at the cylinder bank with the manual pull lever.

9.5.1.2.4.7 Carbon Dioxide Suppression Systems

9.5.1.2.4.7.1 Diesel Generator Rooms

A high pressure carbon dioxide fire extinguishing system (Cardox System) is provided for protection of the diesel generator rooms. The system includes the following principal features:

a) Automatic detection and release by heat actuated devices

b) Remote manual actuation by a pneumatic release that operates directly on pilot discharge heads and is entirely independent of automatic releases

c) Direct manual actuation at the cylinder bank

d) Hazard alarm

e) Alarm for personnel exit with discharge delayed for twenty seconds after automatic release only

f) Pressure switch shutdown of electrical equipment, and

g) Pressure release for fuel supply valves.

While not physically connected or dependent on any other system, the FDAS complements the CO_2 Suppression System by providing remote annunciation at the FAP in the Control Room and closing the appropriate fire dampers via the FDAP.

The Cardox system is a multiple hazard system. Both banks of CO_2 bottles (nineteen 75 lb bottles total) will discharge into the room containing the fire. The system is released automatically if the temperature in the hazard trips the heat actuated device. Closing of the normally open switch element in a thermostat will cause opening of a normally deenergized, normally closed solenoid valve to admit control pressure to the pilot cylinder discharge heads.

A 50 lb bottle of CO_2 is provided to operate the CO_2 whistle alarm which annunciates the discharge in the room. The system includes pressure trips to release the fire door weights, thereby allowing the door to roll closed, if open. Electrical signals also cut off the fuel oil supply and shut off the exhaust fan. The heating, ventilating and air conditioning (HVAC) ducts are equipped with fused link fire dampers, which allow the dampers to close on high temperature. Redundant signals from FDAS provide automatic damper closure on initiation of gas flow.

The system design is consistent with NFPA 12 and the particular considerations of the guideline.

9.5.1.2.4.7.2 North and South Cable Vaults

The North and South Cable Vaults high pressure carbon dioxide fire suppression system is operated by electro-pneumatic control systems. It is completely independent of all other fire suppression systems. Electrical signals are used to operate pilot valves and dampers and to sound alarms. Pneumatic

pressure is used to operate valves. The CO_2 suppression system can be actuated by the following methods:

a) Automatically, utilizing

1) Low Voltage Ionization Detectors
2) Thermal Fire Detectors
3) Photo-Electric Smoke Detectors

b) Remote-Manual Actuation

1) FADP-A1
2) FDAP-B1
3) Manual-Pull Stations
   (a) Inside North Cable Vault (Alarm Only)
   (b) Inside South Cable Vault (Alarm Only)
   (c) Outside South Cable Vault by FD 6

c) Direct manual actuation, which uses handwheels on the pilot operated discharge heads.

The FDAS complements the CO_2 suppression system by providing remote annunciation at the FAP in the Control Room and at the appropriate FDAP. The FDAP also activates the local zone evacuation alarm and closes the appropriate fire dampers.

The CO_2 suppression system is electrically interconnected to the FDAS and is dependent on the FDAS for the automatic and remote-manual actuation signal.

9.5.1.2.4.8 Portable Extinguishers

Portable extinguishers are provided at strategic locations throughout the plant. Hand CO_2 extinguishers are located in the Control Room, cable vaults and electrical equipment rooms. Portable CO_2 extinguishers are also provided inside containment. The Auxiliary Building is also provided with CO_2 and dry chemical, portable fire extinguishers in order to minimize the spread of radioactive contamination in the event of fire. Dry chemical extinguishers are also available for fire protection for the Fuel Handling Building. Portable extinguishers are also provided at locations throughout the Turbine Building. Large wheeled extinguishers are also present in the Turbine Building and the ground floor of the Auxiliary Building. The provision of these extinguishers is consistent with NFPA 10 and the considerations of the guideline.

9.5.1.2.4.9 Fire Protection and Extinguishing Systems for the Control Room and Other Operating Areas Containing Safety-Related Equipment and Cables

9.5.1.2.4.9.1 Control Room

The Robinson plant Control Room is located on the third level of the Auxiliary Building, separated from the remainder of the plant. The layout is provided on Figure 9.5.1-12. To restrict the possibility of fire originating in the Control Room, no combustible material, trim or furnishings are used in its construction.

Electrical circuits are limited to those associated with lighting, instrumentation and control. Lighting circuits are 120 volt; instrumentation and control circuits are either 120 volts AC, 125 volts DC or at the millivolt level. All 120 and 125 volt circuits are protected against both overload and short circuits by either fuses or circuit breakers. The power levels on the millivolt circuits are so low that it is inconceivable that short circuits in these could become a fire hazard.

All lighting wiring is either in steel conduits or enclosed in metal wireways built into the lighting fixtures. All instrumentation and control wiring is inside the panels or control boards in which the wires are terminated. Switchboard wiring is flame resistant.

Because of the overload and short circuit protection provided for the circuits in the Control Room, and because all electrical wiring and devices are surrounded by or mounted in metal enclosures, the fire hazard presented by the electrical equipment due to either electrical faults, or an externally applied flame, is considered minimal.

Safety-related cabinets in the Control Room have smoke detectors placed inside the cabinets for prompt detection of fire. Figure 9.5.1-12 shows the location of the smoke detectors. Cabinets A, B, C, D, and E are actually a single, undivided cabinet. Three smoke detectors are placed in this cabinet; one in Sections A and B, one in Section C, and one in Section D and E. Other cabinets do not require detectors since they are either not related to safe shutdown, or redundancy precludes the requirement.

In the event a fire does occur in the Control Room, the operator has available CO_2 fire extinguishers sized and located in accordance with NFPA specification. A manual hose station on the Turbine Building mezzanine could possibly be used if water were required to suppress a fire in the Control Room. A hose station in the Hagan Room is also provided for the Control Room fires.

A Control Room fire getting out of control and requiring abandoning of the Control Room is not considered credible due to the noncombustible and flame resistant nature of the building and electrical system. However, even should such a situation arise and the operators were forced to leave the Control Room, individual controls (local and at the motor control centers) are available for safely shutting down the plant, if required. Also, a dedicated shutdown system exists (Section 7.4.1.2) which will bring the plant to a safe shutdown condition in the event of a fire in the Control Room or other key areas.

Design features also exist for protection against the spread of fire into the Control Room from other areas. All control wiring entering the Control Room does so via slots in the Control Room floor. The slots are located beneath the control or instrumentation panels in which the wires are terminated. To prevent flames and products of combustion from a cable spread room fire entering the Control Room, each slot is sealed with a plate fabricated and drilled so that cables pass through individual holes in the plate and are individually sealed and supported. To provide additional separation from the relay room on the control level, a three hour rated fire door is provided.

The Control Room is air conditioned by its own air conditioning unit located in the equipment room below. The air intake is ducted from a louver in the exterior wall. If a remotely located fire causes smoke to be drawn into the Control Room, controls are available in the Control Room to operate motor-operated dampers. These place the air conditioner in a recirculating mode with filters in the circuit to clear the control room of entrapped smoke. The recirculating mode can also be used to clear the air if smoke is generated in the Control Room. Self-contained breathing units are also available in the Control Room locker if required.

Based on the above features, the Control Room conforms to the guidelines.

9.5.1.2.4.9.2 Cable Spreading Room

The Unit 2 cable spread room is an area with significant concentrations of cabling for safety-related equipment and systems including those required for safe shutdown. All cables entering the Unit 2 cable spread room do so via trays which penetrate the wall between the cable spread room and the electrical equipment room. The penetrations are through windows cast in the wall just large enough to allow passage of the trays. The air spaces around the trays and cables are sealed by field fitted aluminum plates, glass wool and a flame retardant mastic protective coating. In both the electrical equipment room and cable vault, cables for mutually redundant safety features systems are run in separate trays. These separate trays run with as much lateral space between them as is physically possible to prevent any single fire or other hazard from disabling all systems.

Cabling in the cable spread room is of PVC construction and divisional cable separation does not meet the guidelines of Regulatory Guide 1.75. To provide additional fire protection and to alleviate the deficiencies of cable construction and separation, all cable trays are coated with a fire retardant. This coating reduces the likelihood of cable fires and the effect of non-cable related fires on safety-related cables.

The cable spread room is ventilated by an air circulating system separate from the Auxiliary Building system. In the event of a fire in this area, the operator has control of the supply fan from the Control Room so that the circulating air can be cut off. The presence of fire in the cable spread room is alarmed in the Control Room by a detection system. The room is constructed without the use of combustible materials, trim or furnishings. Manual hose stations are available nearby from the electrical equipment room and Turbine Building mezzanine.

The cable spread room is adequately isolated from the remainder of the plant by its present design. Reinforced concrete walls, ceiling, and floor provide the primary barriers. Additionally, three hour rated fire doors are provided at the exterior and interior entrance to the room. These doors are widely separated to provide personnel access and are within the combustible loading of the room. Cable trays are generally overhead, which provide adequate provision for access also. Electrical penetrations as described above are also of adequate design.

Because of the importance of this room for safe shutdown purposes, additional features for fire protection increase the capability for safe shutdown even if the room should be lost to fire. These features provide a defense in depth approach consistent with the overall criteria of the fire protection program at the Robinson plant.

An automatic total flooding Halon suppression system described in Section 9.5.1.2.4.6 is provided for the Cable Spread Room. The ventilation system fire dampers automatically close on initiation of gas flow. A fixed water suppression system was not considered appropriate due to the presence of important electrical relay racks, difficulty of arranging a system that would wet all cable tray runs and possible drainage problems. To provide additional protection, however, by way of safe shutdown capability, alternate cable routings independent of this area for necessary equipment are provided. This safe shutdown capability is discussed in more detail in Section 9.5.1.3.2. The degree of fire protection is provided for the Cable Spreading Room consistent with the overall evaluation criteria and the stated guidelines.

9.5.1.2.4.9.3 Emergency Switchgear Rooms

Safety-related switchgear, inverters and relay racks are located in the Auxiliary Building adjacent to the cable spreading room. The portion of this area containing the seal water injection tank is fenced off from the remainder of the area for access control. Three-hour rated fire doors are installed to provide the separation from the remainder of the plant. An automatic fire detection system is installed in this area as are portable extinguishers and a manual hose station with an electrically safe nozzle.

Cabling in this area is of PVC construction. To provide additional fire protection, all cable trays are coated with a fire retardant. This reduces possible spread of fire along cable trays or to redundant divisions. A total flooding halon suppression system is installed for this area which contains critical emergency buses. Ventilation system fire dampers are automatically actuated to close upon initiation of gas flow. To provide additional capability for safe shutdown, alternate cable routings and power supplies independent and separate from this area are provided. Capability for power transfer to an alternate supply outside this area, as described in Section 9.5.1.3.3, will assure safe shutdown even in the event of loss of this area.

The protective features of this area provide fire protection consistent with the overall evaluation criteria and the stated guidelines.

9.5.1.2.4.9.4 Remote Safety-Related Panels

The remote safety-related panels presently installed at the Robinson plant consist of the following:

a) Waste disposal boron recycle panel

b) Waste evaporator panel

c) Remote shutdown equipment panels, and

d) Diesel generator local control panels.

The first two panels are located in the lower hallway of the Auxiliary Building. Fire detection provided for this area is due to the presence of safety-related cabling. The remote shutdown panels are located throughout the plant and are normally in the same room as the equipment controlled by them. Two sets of remote shutdown panels that control equipment not located in their general area are protected by existing fire detection systems (electrical equipment room and rod control power supply room). Fire detection systems for safety-related rooms containing these controls are discussed in Section 9.5.1.2.4.5. Fire detection and suppression provided to the diesel room local control panels are discussed in Section 9.5.1.2.4.7.1.

Manual hose stations and portable fire extinguishers are available for fire protection in all areas containing remote safety-related panels; and as indicated above or discussed for specific rooms, automatic fire suppression is provided for several of the areas containing such panels or controls. Even in the event of fire and loss of the waste disposal boron recycle panel or waste evaporator panel, safe shutdown is not compromised nor will radioactive release result. Based on all these considerations, more than adequate compliance with the guidelines exists.

9.5.1.2.4.10 Fire Protection System Power Sources

Primary and backup fire protection is provided by a variety of methods, depending on characteristics of the area, function of contained equipment, effect of loss of function, etc. These fall into the following categories:

a) Areas with gas agent primary suppression systems and manual methods (hose and portable extinguishers) as backup

b) Areas with sprinkler systems and manual methods as backup, and

c) Areas dependent entirely on manual suppression.

Portable extinguishers and manual hoses are the primary and backup suppression methods in these areas.

Power is provided to fire detection systems through essential electrical power buses.

In the areas using gas agent suppression, cross-zoned detectors are provided. The two detector zones in each area are independent of each other (including all system components and power supplies) so that no single failure would result in loss of detection capability. Further, because of the independence of the gas agent and water (manual) suppression systems, no single failure would cause loss of both.

Detection is provided in safety-related areas which have sprinklers as the primary fire suppression means. In the event of a single failure affecting the sprinklers, the detection system would not be lost due to the failure, and manual suppression (the backup) could be used. The fire water system, with the exception of the containment system, is designed so that no single failure will result in loss of capability of both sprinklers and manual hoses. To achieve this, water supplies to sprinklers and hoses are taken from different

points on the fire main loop and appropriate isolation valves provided. The containment fire water system enters through a single penetration sleeve and has a single supply line outside containment.

For the remaining areas, no single failure will result in loss of both portable extinguishers and manual hose capability.

For additional information on specific power sources for fire detection and actuation instrumentation on a fire zone by fire zone basis, see Table 9.5.1-3.

9.5.1.2.4.11 Electric Cable Construction, Cable Trays, and Cable Penetrations

Details for cables are:

a) Cable trays are entirely of metal construction and present no combustible hazard, therefore meeting the guideline.

b) See Section 9.5.1.2.4.9.2 for discussions relating to the cable spread room.

c) Safety-related cable trays outside the cable spread room have been evaluated for fire protection provisions. Since safety-related cable runs in the Auxiliary Building did not satisfy the requirements of Regulatory Guide 1.75 and consist primarily of PVC jacketed cables, application of a flame retardant mastic coating was provided for trays containing engineered safeguards cable. Cabling inside containment has silicon rubber jacket material which has superior fire resistant properties compared to the PVC cable. Automatic water sprinklers protect safety-related cable in the hallway of the Auxiliary Building ground floor near the station air compressors. No critical equipment requiring protection from water damage is in this area. This and other areas have manual hose stations and fire extinguishers for additional protection. A cable tray fire would be a rather slowly propagating fire (1-2 in./min). The flame retardant coating and automatic detection provide adequate protection unless there is a significant exposure fire hazard. Based on the fire hazards analysis and alternate methods of fire suppression in specific areas (see Section 9.5.1.2.4.9), automatic water sprinkler systems are not considered warranted for all cable trays outside the cable spread room.

Existing features provide protection in line with the criteria of the fire safety evaluation.

d) Cable and cable tray penetrations of fire barriers have been sealed to give adequate fire resistance. Cables which enter the cable spreading room from the electrical equipment room do so via trays which pass through openings cast in the wall just large enough to allow tray passage. The air spaces around the trays and cables are sealed by field fitted aluminum plates, glass wool and flame retardant mastic coating. Control wiring entering the Control Room from the cable spreading room uses slots in the floor. Each slot is sealed with a plate; the fabricated and drilled so that cables pass through individual holes in the plate; the cables are individually sealed and supported. In other locations where electrical cable trays penetrate the walls and floors, they are packed with glass wool and protective flame retardant mastic coating.

Sketches describing cable penetrations for walls and floors are given in Figures 8.3.3-1, 8.3.3-2, and 9.5.1-7. The penetration design is similar to that recommended in the International Guidelines for Fire Protection of Nuclear Power Plants (1974). With this construction and use of flame retardant coating, the intent of this guideline is met.

e) Prior to the selection of cabling to be used in the plant, an extensive flame testing program took place which included ASTM vertical flame testing and bonfire tests. Cables were specified on the basis of results from these tests. IEEE-383 testing was not applicable at the time of these tests, but the following tests were made to determine the flame resistant qualities of various cable coverings and insulations:

1) Standard Vertical Flame Test - made in accordance with ASTM-D-470-59T. "Tests for Rubber and Thermalplastic Insulated Wire and Cable."

2) Five-Minute Vertical Flame Test - made with cable held in vertical position and 1750`F flame applied for 5 min.

3) Bon-Fire Test - consisting of exposing, for 5 min, bundles of three or six cables to flame produced by igniting transformer oil in a 12-in. pail. The cable was supported horizontally over the center of the pail. The time to ignite the cable and the time the cable continued to flame after the fire was extinguished were noted.

Engineered safeguards cable trays containing cable with PVC jackets are covered with a flame retardant coating. Thus, the intent of the guideline is met.

f) For new cable installations, cable construction that does not give off corrosive gases while burning will be considered to the extent practicable in compliance with this guideline.

g) No miscellaneous storage in cable trays or raceways is allowed. All trays are maintained free of debris or other hazardous items. Control of cable tray, conduit, etc., is maintained administratively.

A hydrogen pipe supplying the volume control tank now runs approximately six inches beneath three safeguards and control cable trays (El) in the lower level of the Auxiliary Building. Piping is marked as containing hydrogen approximately every 10 ft. The pipe material is Schedule 40 stainless steel; therefore, no corrosion problem could result in leakage. Hydrogen pressure is maintained at 130 psig max.

h) Smoke venting for the cable spread room would not be appropriate for the halon suppression system. Smoke could be vented by opening the door to exterior. These capabilities provide adequate compliance for this guideline as appropriate to the area.

i) In general, the Control Room contains a minimum of cables, which enter the Control Room from the cable spreading room below and terminate close by. There are no cable trenches or culverts in the Control Room.

9.5.1.3 Safety Evaluation (Fire Hazards Analysis)

9.5.1.3.1 Method of Evaluation

9.5.1.3.1.1 Methodology

The evaluation of the fire protection program of the Robinson plant was carried out by the combined performance of a fire hazards analysis and a comparison of the existing provisions against the guidelines of Appendix A to BTP APCSB 9.5-1. Where deficiencies with Appendix A were noted, the disposition of the deficiency was based on the fire hazards analysis and the criteria for the overall evaluation. A more detailed description of the methodology is given in the following step-by-step summary.

a) Plant design features relating to fire safety were determined. These include overall plant layout, type of location of combustible materials, type of construction used and its fire resistance characteristics, fire detection and fire suppression systems, separation criteria used, etc.

b) Areas containing equipment and components important to safety were identified. These areas and adjacent areas with possible fire hazard effects were subdivided into fire areas on the basis of existing walls and other boundary fire barriers. For each fire area, the following were determined:

1) Total fire load (Btu/sq ft) in the area assuming total combustion of cable insulation, oil, and other combustibles

2) Major equipment or systems in the area

3) Fire detection and suppression systems

4) Adequacy of the fire barrier boundaries

c) Components and systems required to safely shut down the reactor and plant, and dissipate residual heat were identified. This included not only major components, but cables, control, power sources, and required auxiliaries such as cooling.

d) Effects of postulated fires on safe-shutdown items were determined, and safe-shutdown capability in the event of fires was evaluated. Consequences with and without active fire protection were considered.

e) For each area, the adequacy of existing fire detection and fire suppression systems was evaluated in the light of the foregoing items, and considering the combustibility of materials, potential ignition sources, and the single failure criterion.

f) Plant features were evaluated which impact directly or indirectly on the plant fire brigade's ability to reach and effectively fight credible fires.

9.5.1.3.3.2 Functional Requirements for Safe Shutdown

In order to provide safe shutdown capability in spite of postulated fires, systems and components required to achieve and maintain a safe shutdown condition must be identified. Those required have been identified in accordance with General Design Criterion 19. The general functional requirements are:

a) To monitor and control the primary system coolant inventory (pressurizer level)

b) To remove decay heat by means of feedwater addition to the steam generators with atmospheric venting of steam

c) To control reactivity by boration of the primary system

d) To monitor reactor neutron level to assure that subcriticality is maintained, and

e) To provide auxiliary services (cooling water and electric power) required by the components which directly perform these functions.

9.5.1.3.3.3 Systems and Components Required

The system and major components required to fulfill these requirements, and the auxiliaries required, are as follows:

a) Primary System Coolant Inventory

    1) Charging Pump

    2) Pressurizer Level Instruments

    3) Chemical and Volume Control System Letdown Valves

    4) Pressurizer Pressure Indicator

b) Decay Heat Control

    1) Auxiliary Feedwater Pump (Steam Driven)

    2) Steam Generator Level

    3) Relief Valves

c) Monitor Reactor Neutron Level - Startup Neutron Channel

d) Auxiliaries

    1) Component Cooling Pump

    2) Service Water Pump

    3) 480V AC Electric Power

    4) 125V DC Electric Power

9.5.1.3.3.4 Shutdown Outside Control Room

In the event of fire in a critical area such as the Control Room, cable spread room, or emergency switchgear area, the capability of achieving safe shutdown is still required, possibly using features not normally used. The Auxiliary Building, its equipment, and furnishings have been designed to minimize the likelihood of fire rendering the Control Room inaccessible. However, provisions have been made for plant operator to shut down and maintain the plant in a safe condition by means of controls located outside the Control Room. During a period of Control Room inaccessibility or loss due to fire in other areas such as the cable spreading room, or emergency equipment room, which could have effects similar to loss of the Control Room, the reactor will be tripped and the plant maintained in the hot shutdown condition. If the period extends for a long time, the Reactor Coolant System (RCS) can be borated to maintain shutdown as xenon decays. Local controls and indicators are provided to accomplish these functions.

If the Control Room should be evacuated suddenly without any action by the operators, the reactor can be tripped by either opening rod control breakers at the reactor trip switchgear or actuating the manual turbine trip at the control station in the Turbine Building. The following systems and equipment are provided to then maintain the plant in a safe shutdown condition from outside the Control Room:

a) Pressurizer Pressure and Level Control

Following a reactor trip, the primary temperature will automatically reduce to the no-load temperature condition as dictated by the steam generator temperature conditions. This reduction in the primary water temperature reduces the primary water volume, and if continued pressure control is to be maintained, primary water makeup is required.

The pressurizer level is controlled in normal circumstances by the chemical and volume control system, which includes the charging pumps. Water may readily be obtained from normal sources, i.e., the volume control tank; alternatively, the refueling water storage tank is another source.

Pressurizer level and pressure indicators are present with one set visible from the auxiliary feedwater pump control panel and one set visible from the charging pump local control panel. All instruments at the auxiliary feed pump control panel are grouped on a local enclosed gauge board.

b) Residual Heat Removal

Following a normal plant shutdown, an automatic steam dump control system bypasses steam to the condenser and maintains the reactor coolant temperature at its no-load value. This implies the continued operation of the steam dump system, condensate circuit, condenser cooling water, feed pumps and steam generator instrumentation. Failure to maintain water supply to the steam generators would result in steam generator dry-out and loss of the secondary system for decay heat removal. Redundancy and full protection where necessary are built into the system to ensure the continued operation of the steam generator units. If the automatic steam dump control system is not available, independently controlled relief valves on each steam generator maintain the

steam pressure. These relief valves are further backed up by coded safety valves on each steam generator. The steam relief facility is adequately protected by redundancy and local protection. For decay heat removal, it is necessary only to maintain the control on one steam generator.

For the continued use of the steam generators for decay heat removal, it is necessary to provide a source of water, a means of delivering that water and, finally, instrumentation for pressure and level indication.

Feedwater may be supplied to the steam generators by the electrical feed pumps or by the steam-driven feed pump; these pumps and associated valves have local controls. Level indication for the individual steam generators is present with one set visible from the auxiliary feedwater pump controls and one set visible from the bypass feed regulator valves. Pressure indication for individual steam generators is visible from the auxiliary feedwater pump controls.

c) Reactivity Control

Following a normal plant shutdown to hot shutdown condition, soluble poison may be added to the primary system to assure subcriticality. For boron addition, the chemical and volume control system is used. Routine boration requires the use of boric acid transfer pumps with tanks and associated piping. The boric acid transfer pumps supply concentrated boric acid to the charging pumps, which then pump it to the RCS.

Note that with the reactor held at hot shutdown conditions, boration of the plant is not required immediately after shutdown. The xenon transient does not decay to the equilibrium level until some 10 to 15 hr after shutdown, and a further period would elapse before the reactivity shutdown margin provided by the full length control rods had been cancelled.

d) Monitor Reactor Neutron Level

A nuclear instrumentation startup channel is required to monitor core activity during the shutdown condition.

9.5.1.3.3.5 Electric Power Supply

For long-term maintenance of safe shutdown conditions, component cooling and service water are also required. Electric power is required to operate the various pumps and systems.

Multiple outside sources of power are available to the plant for both normal operation and shutdown functions. Normal operations utilize both outside and unit-generated power. Separation of these two sources is maintained in the 4160 V, 480 V, and lower voltage systems (see Figure 8.1.2-1). The plant auxiliary equipment is arranged electrically so that redundant items receive their power from the two different sources. Two of the three charging pumps are supplied from the 480 V Buses E1 and E2. Two of the three component cooling pumps are supplied from the 480 V Buses E1 and E2. Two of the four service water pumps are supplied from 480 V emergency Bus E1 and two pumps from Bus E2.

A 480 V dedicated shutdown bus supplied from offsite power or from a diesel generator provides a primary power feed to one component cooling pump, one charging pump and an alternate feed to one service water pump, to MCC5, and other essential loads. (See Figure 8.3.1-4)

Refer to Chapter 8 for a detailed description of the electrical power system.

9.5.1.3.4 Fire Area Safety Analysis

This section provides an area-by-area analysis of the fire hazards and potential consequences for fire areas of the Robinson plant. The fire zones are identified on Figures 9.5.1-2 through 9.5.1-6. For each area or zone, the major equipment contained, and their safety function, are given. Installed and proposed fire protection features are provided and design features relating to fire safety of the area. An inventory of combustible material is also given including fire load and maximum fire severity in terms of time of duration as determined from Table 6-8A of the Fire Protection Handbook (Fourteenth Edition). The possible consequence of a design basis fire in each area is discussed, based on worst case conditions.

This section is an update of the Fire Protection Program Review for HBR 2 issued January 1, 1977.

9.5.1.3.4.1 Zone 1 - Diesel Generator "B" Room

a) ACCESS

Zone 1, the Diesel Generator "B" Room is located on the ground floor of the Auxiliary Building on the 226.0 ft elevation. Access to this area is through the emergency diesel door, Fire Door 23.

b) COMBUSTIBLES

Combustible material located in Zone 1 consists of 200 gal of diesel oil in the day tank with a fire load of 49,300 Btu/ft^2. In addition, there are 250 gal of lubricating oil located in Zone 1 with a fire load of 61,600 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 1 is 83 minutes.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 1 consists of two ultraviolet flame detectors, one heat detector, and one manual pull station. Any one of these detectors will, through the Fire Detection and Actuation Panels (FDAP-A1 and B1):

1) Actuate the Fire Alarm Panel (FAP) in the Control Room

2) Close Fire Dampers 1, 2, 3, 75, and Fire Door 23, and

3) Stop the Fuel Oil Transfer Pump "B" and Zone 1 intake and exhaust fans HVS-5 and HVE-17.

Additional fire detection in Zone 1 consists of two heat detectors. These detectors will, through the CO_2 suppression system:

1) Actuate the CO_2 system

2) Close Fire Door 23, and

3) Stop the Fuel Oil Transfer Pump "B" and Zone 1 intake and exhaust fans HVS-5 and HVE-17.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 1 consists of an automatic high pressure carbon dioxide (CO_2) system. This system is actuated:

1) Automatically by Heat Actuated Devices (HAD)

2) Manually by activating the "B" Diesel Generator remote manual release located on the outside of the west wall of Diesel Generator "B" room, and

3) Manually by first opening the valve on either CO_2 bottle B9 or B8 and then opening the "B" Diesel General pilot control valve. This valve is located approximately 6 ft from the floor just to the left of the bottle rack.

This system, only when actuated automatically, provides a 25 sec delay before discharging. There is no means to isolate this system after actuation.

The following portable fire suppression equipment is available for use in Zone 1:

1) One 10 lb Dry Chemical Fire Extinguisher (Zone 1)

2) One 20 lb CO_2 Fire Extinguisher (Zone 1), and

3) One Hose Station (Zone 11).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 1 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, there is an automatic closing three-hour rated fire door (FD-23) which closes upon actuation of the CO_2 system or the low voltage fire detection system.

f) SHUTDOWN CAPABILITY

Assuming worst case in which a fire in Zone 1 totally disables Diesel Generator "B" and all Diesel Generator "B" auxiliary equipment, this incident does not reduce plant shutdown capability. Diesel Generator "B" is the emergency power supply to 480V Emergency Bus No. E2. Normal power source to this bus is Station Service Transformer "C" which is normally energized. There is a backup energy source to Bus E2 via manually closed tie breakers for Bus E1 and E2. However, this requires bypassing the position sensitive interlocks or the E1-E2 tie breakers. If the event were to disable Emergency Bus E2, sufficient redundant equipment is powered from Emergency Bus E1 to safely shut down the plant.

9.5.1.3.4.2 Zone 2 - Diesel Generator "A" Room

a) ACCESS

Zone 2, the Diesel Generator "A" Room, is located on the ground floor of the Auxiliary Building on the 226.0 ft elevation. Access to this area is through the Diesel Generator Room door, Fire Door 24.

b) COMBUSTIBLES

Combustible material located in Zone 2 consists of 200 gal of diesel oil located in the day tank with a fire load of 49,300 Btu/ft^2. In addition, there are 250 gal of lubricating oil located in Zone 2 with a fire load of 61,600 Btu/ft^2. The maximum fire severity for both combustibles located

in Zone 2 is 33 minutes. Zone 2 also contains a small amount of miscellaneous combustibles stored at the south end of the room, adding less than 5,000 Btu/ft^2 to the fire load.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 2 consists of two ultraviolet flame detectors, one heat detector, and one manual pull station. Any one of these detectors will, through the FDAP-A1 and B1:

1) Actuate the FAP in the Control Room

2) Close Fire Dampers 4, 5, 6, 7, and Fire Door 24, and

3) Stop the Fuel Oil Transfer Pump "A" and Zone 2 intake and exhaust fans HVS-6 and HVE-18.

Additional fire detection in Zone 2 consists of two heat detectors. These detectors will, through the CO_2 suppression system:

1) Actuate the CO_2 system

2) Close Fire Door 24, and

3) Stop the Fuel Oil Transfer Pump "A" and Zone 2 intake and exhaust fans HVS-6 and HVE-18.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 2 consists of an automatic high pressure carbon dioxide (CO_2) system. This system is actuated:

1) Automatically by HAD

2) Manually by activating the "A" Diesel Generator remote manual release, located on the outside of the west wall of Diesel Generator "B" room, and

3) Manually by first opening the valve on either CO_2 bottle B9 or B8 and then opening the "A" Diesel Generator pilot control valve. This valve is located approximately 6 ft from the floor just to the left of the bottle rack.

This system, only when actuated automatically, provides a 25 sec delay before discharging. There is no means to isolate this system after actuation.

The following portable fire suppression equipment is available for use in Zone 2:

1) One 10 lb Dry Chemical Fire Extinguisher (Zone 2)

2) One 20 lb CO_2 Fire Extinguisher (Zone 2), and

3) One Hose Station (Zone 11).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 2 are of noncombustible fire resistant materials. The walls and floors are of reinforced concrete and provide a three-hour rated fire barrier. In addition, there is an automatic closing three-hour rated fire door (No. 24) which closes upon activation of the Cardox System or the low voltage fire detection system.

f) SHUTDOWN CAPABILITY

Assuming worst case in which a fire in Zone 2 totally disables Diesel Generator "A" and all Diesel Generator "A" auxiliary equipment, this incident does not reduce plant shutdown capability. Diesel Generator "A" is the emergency power supply to 480V Emergency Bus No. E1. Normal power source to this bus is Station Service Transformer "A" which is normally energized. There is a backup energy source to Bus E2 via manually closed tie breakers for Bus E1 and E2. However, this requires bypassing the position-sensitive interlocks on the E1-E2 tie breakers. If the event were to disable Emergency Bus E1, sufficient redundant equipment is powered from Emergency Bus E2 to safely shutdown the plant. A fire in the Zone provides potential for a loss of "B" Diesel Generator because the service water supply line runs in the overhead of this room.

If the latter case occurs, a safe plant shutdown can be accomplished by use of the Dedicated Shutdown System.

9.5.1.3.4.3 Zone 3 - Safety Injection Pump Room

a) ACCESS

Zone 3, the Safety Injection Pump Room, is located on the ground floor of the Auxiliary Building on the 226.0 ft elevation. Access to this area is through Fire Door 1 or Security Door 27.

b) COMBUSTIBLES

Combustible material located in Zone 3 consists of 69 ft^3 of cable insulation with a fire load of 41,600 Btu/ft^2. In addition, there is less than ten gallons of lubricating oil with a fire load of less than 1,695 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 3 is 32 minutes.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 3 consists of two heat detectors, two ionization smoke detectors and two manual pull stations. These detectors will, through the FDAP A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 3. However, the following portable fire suppression equipment is available:

1) Two 20 lb Dry Chemical Fire Extinguishers (Zone 3)

2) One 200 lb Dry Chemical Fire Extinguisher (Zone 12)

3) One 20 lb CO_2 Fire Extinguisher (Zone 11), and

4) One Hose Station (Zone 11).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 3 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, the fire door also provides a three-hour fire rated barrier.

f) SHUTDOWN CAPABILITY

Safety-related equipment located in Zone 3 includes the three Safety Injection Pumps and Motor-operated Valves (MOV) the two Containment Spray Pumps and MOV, the Primary Water Pumps and MOV. A fire in this zone would disable all of these pumps but a safe shutdown of the plant can be accomplished without these pumps.

9.5.1.3.4.4 Zone 4 - Charging Pump Room

a) ACCESS

Zone 4, the Charging Pump Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is through Fire Door 2.

b) COMBUSTIBLES

Combustible material located in Zone 4 consists of 40 gal of lubricating oil in each pump. The fire load of this combustible material is 21,400 Btu/ft^2 with a maximum fire severity of 16 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 4 consists of two heat detectors, two ionization smoke detectors, and one manual pull station. These detectors will, through the FDAP A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 4. However, the following portable fire suppression equipment is available:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 13)

2) Two 20 lb CO_2 Fire Extinguishers (Zone 13), and

3) One Hose Station (Zone 13).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type, (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 4 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, the fire door leading into the Zone also provides a three-hour fire rated barrier.

f) SHUTDOWN CAPABILITY

The following safety-related shutdown equipment is located in Zone 4. A, B, and C Charging Pumps and associated valving, and the Remote Shutdown Panel are used for shutdown monitoring when the Control Room is inaccessible. A fire in this Zone could result in the loss of all three Charging Pumps and the remote Shutdown Panel. A loss of all three Charging Pumps will have an adverse effect on shutting down the plant by inhibiting easy volumetric control of the Reactor Coolant System (RCS). This will not prevent the ability to shutdown the plant. An alternate supply of water to maintain RCS inventory can be provided from the three High Head Safety Injection Pumps which take suction on the Refueling Water Storage Tank. This method will require a reduction in RCS pressure to a pressure less than the shutoff head of the Safety Injection Pumps. Extra precautions should be taken in this mode to insure that sufficient subcooling is maintained in the RCS. Reactor Coolant Pump seal parameters should be closely monitored if all three Charging Pumps fail.

9.5.1.3.4.5 Zone 5 - Component Cooling Pump Room

a) ACCESS

Zone 5, the Component Cooling Pump Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is through either Fire Door 3 or Security Door 24.

b) COMBUSTIBLES

Combustible material located in Zone 5 consists of 66 ft^3 of cable insulation. The fire load of this combustible material is 15,600 Btu/ft^2 with a maximum fire severity of twelve minutes.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 5 consists of three heat detectors, three ionization smoke detectors, and one manual pull station. These detectors will, through the FDAP-A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 5. However, the following portable fire suppression equipment is available:

1) Two 20 lb CO_2 Fire Extinguishers (Zone 13)

2) One 20 lb Dry Chemical Fire Extinguisher (Zone 13), and

3) One Hose Station (Zone 13).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 5 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, Fire Door 3 provides a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 5 includes the following safety-related equipment: A, B, and C Component Cooling Water (CCW) Pumps, both CCW Heat Exchangers, both Boric Acid Tanks and the Transfer Pumps with associated instruments and heat tracing, the transfer switches for "D" Service Water Pump, and the transfer switch for the North Service Water Header Discharge Valve (V6-12D).

There is a fire barrier wall, for physical separation, located between CCW Pumps "A" and "B". This separation provides protection from a total loss of component cooling water flow if a major fire occurs in Zone 5.

A major fire in this Zone would result in a loss of power supply to "D" Service Water Pump. This will not adversely effect plant shutdown capability with the remaining 3 service water pumps still available.

Another effect seen on the Service Water System would be a loss of operation of the North Header Discharge Valve (V6-12D). Shutdown capability is not inhibited because this valve is normally open and the north and south headers are normally cross-tied.

Loss of the other equipment including Boric Acid Transfer Pumps will not prevent a safe plant shutdown.

9.5.1.3.4.6 Zone 6 - Hot Lab and Counting Room

a) ACCESS

Zone 6, the Hot Chemical Lab and Counting Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is through Fire Door 4.

b) COMBUSTIBLES

Combustible material located in Zone 6 consists of miscellaneous supplies and laboratory chemicals. The fire load of this combustible material is less than 40,000 Btu/ft^2 (estimated) with a maximum fire severity of less than 30 min. A flammable storage cabinet is located in this zone for storage of combustible material.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 6 consists of one heat detector, one ionization smoke detector, and one manual pull station. These detectors will, through the FDAP A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 6. However, the following portable fire suppression equipment is available:

1) One 10 lb Dry Chemical Fire Extinguisher (Zone 6)

2) One 20 lb CO_2 Fire Extinguisher (Hot Lab, Secondary Chemistry Side)

3) One 20 lb Dry Chemical Fire Extinguisher (Zone 13)

4) Two CO_2 Fire Extinguishers (Zone 13), and

5) One Hose Station (Zone 13).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 6 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, Fire Door 4 provides a three-hour fire rated barrier.

f) ROOM VENTILATION

Room ventilation for Zone 6 is provided by the Auxiliary Building HVAC system. This system supplies air to this zone from supply fan HVS-1. Air is

drawn through the area and exhausted to the stack through exhaust fans HVE-2A and HVE-2B. Smoke removal from this area may be accomplished through the use of this system, providing that affected Fire Dampers (76, 77, and 78) have been reset properly.

NOTE: Combustion products may build up quickly on the absolute filters associated with exhaust fans HVE-2A and HVE-2B, substantially reducing the air flow rate. Portable smoke removal equipment, venting to the outside through Security Door 34 may be a preferable method of smoke removal. Airborne contamination levels should be monitored before and during venting operations.

g) SHUTDOWN CAPABILITY

Zone 6, which is the Hot Chemical Lab and Counting Room, contains no safety-related shutdown equipment. The cable running through these rooms is not safety-related so there would be no effect on shutdown capability of the plant.

9.5.1.3.4.7 Zone 7 - Auxiliary Feedwater Pump Room

a) ACCESS

Zone 7, the Auxiliary Feedwater Pump Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is through Fire Door 5. Emergency access may be obtained through Fire Door 26.

b) COMBUSTIBLES

Combustible material located in Zone 7 consists of two gallons each of lubricating oil in the two pumps, with a fire load of 2,550 Btu/ft^2. In addition, there is 17 ft^3 of cable insulation with a fire load of 38,900 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 7 is 31 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 7 consist of one heat detector, one ionization smoke detector, and one manual pull station. These detectors will, through the FDAP A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 7. However, the following portable fire suppression equipment is available:

1) One 20 lb CO_2 Fire Extinguisher (Entrance to Auxiliary Feedwater Pump Area)

2) One Hose Station (NW of Records Office)

3) One 20 lb Dry Chemical Fire Extinguisher (Health Physics Records Office), and

4) One 20 lb CO_2 Fire Extinguisher (NE corner of condenser).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type, (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 7 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, Fire Doors 5 and 26 in Zone 7 provide a three-hour fire rated barrier.

f) SHUTDOWN CAPABILITY

Safety-related equipment in Zone 7 includes the two motor-driven Auxiliary Feedwater Pumps and Discharge MOV. A fire in this Zone would disable both of the Motor Driven Auxiliary Feedwater Pumps. Safe shutdown can still be accomplished without these two pumps. Steam Generator Water inventory can be maintained by the Normal Feedwater Pumps or, if they are unavailable, by operation of the Steam Driven Auxiliary Feedwater Pump.

9.5.1.3.4.8 Zone 8 - Boron Injection Tank Room

a) ACCESS

Zone 8, the Boron Injection Tank Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is through Security Door 28.

b) COMBUSTIBLES

Combustible material located in Zone 8 consists of 240 lb of charcoal. The fire load of this combustible material is 8,000 Btu/ft^2 with a maximum fire severity of 6 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 8 consists of one heat detector, one smoke detector, and one manual pull station. These detectors will, through the FDAP-A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 8. However, the following portable fire suppression equipment is available:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 8)

2) One Hose Station (Zone 27 Area), and

3) One 20 lb Dry Chemical Fire Extinguisher (Zone 27 Area).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 8 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Safety-related equipment in Zone 8 includes the Boric Acid Injection Tank and the inlet and outlet MOV. This equipment is not essential for a plant shutdown so a fire in this zone does not inhibit a safe plant shutdown.

9.5.1.3.4.9 Zone 9 - North Cable Vault

a) ACCESS

Zone 9, the North Cable Vault Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is through either Fire Door 8A (east door to North Cable Vault) or Fire Door 8B (west door to North Cable Vault).

b) COMBUSTIBLES

Combustible material located in Zone 9 consists of 49 ft^3 of cable insulation. The fire load of this combustible material is 159,000 Btu/ft^2 with a maximum fire severity of 119 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 9 consists of two photoelectric smoke detectors, one ionization smoke detector, one heat detector, and one manual pull station. These detectors will, through the FDAP-A1 and B1:

1) Activate the FAP in the Control Room

2) Actuate the CO_2 suppression system upon receiving a signal from at least one detector on train A and one detector on train B, and

3) Close Fire Dampers 22 and 23 upon receiving a signal from at least one detector on train B.

An additional high voltage detection system is also available in Zone 9 which consists of one heat detector and one ionization smoke detector which when actuated annunciate on the RTGB in the Control Room.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 9 consists of a high pressure carbon dioxide (CO_2) system. This system is actuated:

1) Automatically through the FDAP-A1 and B1 which requires a detection signal from at least one detector to FDAP-A1 and one detector to FDAP-B1.

2) Manually from either FDAP-A1 (located just south of Instrument Air and Service Air Compressor in Zone 12) or FDAP-B1 (located across from the Hot Lab in Zone 13) by switching the zone suppression manual actuation switch to the "Operated" position.

3) Manually from the manual actuation stations (located outside the South Cable Vault in the Dress Out Area) by pulling the train A station and the train B station for Zone 9.

4) Manually from the CO_2 storage area (located in the Pipe Alley, Zone 28) by pulling the safety pins and rotating the handles of the Pilot Control Valves (CD-2, CD-5) counterclockwise until they stop. Next pull the safety pins and rotate handles on the Pilot Discharge Heads (CD-14, CD-16 for main bank or CD-22, CD-24 on reserve bank) counterclockwise until they stop.

This system provides for a fifteen second delay except when activated manually from the CO_2 storage area.

This carbon dioxide system can be isolated by closing ball valves CD-1 and CD-4 (located in the Pipe Alley, Zone 28), thereby restricting the flow of Carbon Dioxide (CO_2). The system cannot be isolated after actuation.

The following portable fire suppression equipment is available for use in Zone 9:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 10), and

2) One Hose Station (Dress Out Area).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 9 are of noncombustible fire material. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. All fire doors in Zone 9 also provide a three-hour fire barrier.

f) SHUTDOWN CAPABILITY

Zone 9 (North Cable Vault) contains the cable runs for the safety-related equipment inside the containment off of train "B". A fire in this Zone would disable train "B" equipment inside the containment. Sufficient redundant equipment is provided from train "A" that safe shutdown capability is not inhibited.

9.5.1.3.4.10 Zone 10 - South Cable Vault

a) ACCESS

Zone 10, the South Cable Vault Room is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this room is through either Fire Door 6 or Fire Door 7.

b) COMBUSTIBLES

Combustible material located in Zone 10 consists of 90 ft^3 of cable insulation. The fire load of this combustible material is 75,300 Btu/ft^2 with a maximum fire severity of 56 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 10 consists of two heat detectors, two ionization smoke detectors, two photoelectric smoke detectors, and one manual pull station. These detectors will, through the FDAP-A1 and B1:

1) Actuate the FAP in the Control Room

2) Actuate the CO_2 suppression system upon receiving a signal from at least one detector on train A and one detector on train B, and

3) Close Fire Dampers 24 and 25 upon receiving a signal from at least one detector on train A and one detector on train B.

An additional high voltage detection system is also available in Zone 10 which consists of two heat detectors and two ionization smoke detectors which when actuated annunciate on the RTGB in the Control Room.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 10 consists of a high pressure carbon dioxide (CO_2) system. This system is actuated:

1) Automatically through the FDAP-A1 and B1 which requires a detection signal from at least one detector to FDAP-A1 and one detector to FDAP-B1.

2) Manually from either FDAP-A1 (located just south of Instrument Air and Service Air Compressor in Zone 12) or FDAP-B1 (located across from the Hot Lab in Zone 13) by switching the zone suppression manual actuation switch for Zone 10 to the "Operated" position.

3) Manually from the manual actuation stations located outside the South Cable Vault in the dress out area by pulling the train A station or the train B station for Zone 10.

4) Manually from the CO_2 storage area (located outside the South Cable Vault in the dress out area) by pulling the safety pins and rotating the handles of the Pilot Control valves (CD-8, CD-11) counterclockwise until they stop. Next pull the safety pins and rotate handles on the Pilot Discharge Heads (CD-18, CD-20 for main bank, or CD-26, CD-28 on reserve bank) counterclockwise until they stop.

The system provides for a fifteen second delay except when activated manually from the CO_2 storage area.

This carbon dioxide system can be isolated by closing ball valves CD-7 and CD-10 (located in the Pipe Alley, Zone 28), thereby restricting the flow of CO_2. The system cannot be isolated during actuation.

The following portable fire suppression equipment is available for use in Zone 10:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 10), and

2) One Hose Station (Dress Out Area).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 10 are of noncombustible fire material. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, all fire doors in Zone 10 provide a three-hour fire barrier.

f) SHUTDOWN CAPABILITY

Zone 10 (South Cable Vault) contains the cable runs for the safety-related equipment inside the containment off of train "A". A fire in this zone would disable train "A" equipment inside the containment. Sufficient redundant equipment is provided from train "B" so that safe shutdown capability is not inhibited.

9.5.1.3.4.11 Zone 11 - Auxiliary Building Hallway/Diesel Generators

a) ACCESS

Zone 11, the Auxiliary Building Hallway/Diesel Generators are located on the ground level of the Auxiliary Building at the 226 ft elevation. Access to this zone from the outside is through Security Door 26. Additional access can be obtained through Fire Door 1 or Zone 12.

b) COMBUSTIBLES

Combustible material located in Zone 11 consists of 134 ft^3 of cable insulation. The fire load of this combustible material is 73,600 Btu/ft^2 with a maximum fire severity of 55 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 11 consists of four ionization smoke detectors, four heat detectors, and two manual pull stations. These detectors will alarm through the FDAP-A1 and B1, and actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system in Zone 11. However, the following portable fire suppression equipment is available:

1) One 20 lb CO_2 Fire Extinguisher (Zone 11)

2) One Hose Station (Zone 11), and

3) One 200 lb Dry Chemical Fire Extinguisher (Zone 12).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 11 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. Three-hour rated fire doors and fire dampers of adjacent fire areas prevent spread of fire to the individual areas. Smoke will propagate into adjacent zones as there are no fire barriers separating Zone 12 from these areas.

f) SHUTDOWN CAPABILITY

Safety-related equipment in Zone 11 includes MCC-5 and cable runs for power and control of both safeguard trains. A fire in Zone 11 would result in the loss of MCC-5 and possibly all three Safety Injection Pumps, both spray pumps, and use of the residual heat removal (RHR) valves.

The loss of this equipment, although serious, will not prevent a safe shutdown of the plant.

Safe shutdown can be accomplished by use of the equipment available from E2 Emergency Bus or by use of the Dedicated Shutdown System.

9.5.1.3.4.12 Zone 12 - Auxiliary Building Hallway With Station And Instrument Air Compressors

a) ACCESS

Zone 12, Auxiliary Building Hallway with Station and Instrument Air Compressor is located on the ground level of the Auxiliary Building. Access to this room from the outside is through Security Door 25. Access from the inside of the Auxiliary Building can be obtained either through Zone 11 or Zone 13.

b) COMBUSTIBLES

Combustible material located in Zone 12 consists of 138 ft^3 of cable insulation with a fire load of 49,000 Btu/ft^2. Additional combustible material consists of less than 10 gal of lubricating oil with a fire load of less than 1,000 Btu/ft^2. The maximum fire severity for both these combustible materials is 38 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 12 consists of three heat detectors, three ionization smoke detectors, six photoelectric smoke detectors, and two manual pull stations. These detectors will, through the FDAP-A1 and B1:

1) Actuate the FAP in the Control Room, and

2) Automatically actuate the preaction water deluge valve after sending a signal from at least one detector on train A and one detector on train B.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 12 consists of a preaction water deluge system which is actuated by:

1) Charging the spray header with water

   a) Automatically through the detection system

   b) Manually at the FDAP-A1 or B-1, and

   c) Manually at the deluge valve.

2) Melting the fused link at individual spray nozzles.

This system may be isolated by manually closing the isolation valve just below the deluge valve. The deluge valve is located against the north wall of Zone 12.

The following portable fire suppression equipment is available for use in Zone 12:

1) One 20 lb CO_2 Fire Extinguisher (Zone 12)

2) One Hose Station (Zone 12)

3) One 200 lb Dry Chemical Fire Extinguisher (Zone 12)

4) One 20 lb CO_2 Fire Extinguisher (Zone 11), and

5) One Hose Station (Zone 11).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type, (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 12 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a

three-hour rated fire barrier. Three-hour rated fire door and fire dampers of adjacent fire areas prevent spread of fire to Zones 2, 4, and 5. Smoke will propagate into adjacent zones as there are no fire barriers separating Zone 11 and Zone 13 from Zone 12.

f) SHUTDOWN CAPABILITY

Safety-related equipment in Zone 12 includes MCC 10 and cable runs for both safeguard trains. A fire in this zone could result in the loss of this equipment.

This will not inhibit the ability to safely shutdown the plant. Plant shutdown can be accomplished without the equipment lost off of MCC 10. MCC 10 is associated with train "A", and with train "B" intact sufficient redundant equipment is available for plant shutdown.

If equipment off train "B" is also lost the plant can be safely shutdown by use of the Dedicated Shutdown System.

9.5.1.3.4.13 Zone 13 - Auxiliary Building Hallway/Component Cooling Room

a) ACCESS

Zone 13, the Auxiliary Building Hallway/Component Cooling Room, is located on the 226 ft elevation of the Auxiliary Building.

Access to this area is through either Fire Door 18, 9, or 4. Access to this area can also be obtained through Zone 12.

b) COMBUSTIBLES

Combustible material located in Zone 13 consists of 120 ft^3 of cable insulation. The fire load for this combustible material is 63,000 Btu/ft^2 with a maximum fire severity of 47 min.

c) FIRE DETECTION

Low voltage fire detection in Zone 13 consists of four heat detectors, four ionization smoke detectors, and one manual pull station. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 13. However, the following portable fire suppression equipment is available:

1) Two 20 lb CO_2 Fire Extinguishers (Zone 13)

2) One 20 lb Dry Chemical Fire Extinguisher (Zone 13), and

3) One Hose Station (Zone 13).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type

(i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 13 are of noncombustible fire resistant material. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. Three-hour rated fire doors and fire dampers of adjacent fire areas prevent spread of fire to Zones 4, 5, 6, 7, and 28. Smoke will propagate into one adjacent zone as there are no fire barriers separating Zone 12 from Zone 13.

f) SHUTDOWN CAPABILITY

Zone 13 contains cable runs for both safeguard trains, cable runs for equipment required for normal shutdown, and the transfer switches for normal and backup power supply to MCC-5.

A fire in this zone could result in the loss of MCC-5, B & C Component Cooling Water Pumps and B & C Charging Pumps, and a possible loss of ability for normal boration of the RCS.

Safe plant shutdown can still be accomplished. "A" Charging Pump, "A" Component Cooling and sufficient auxiliaries from "B" train are available for shutdown of the plant. "A" CCW Pump will be available since its normal supply is the dedicated shutdown bus. "A" Charging Pump is supplied from the dedicated shutdown bus. The loss of MCC-5 will be mitigated by the availability of MCC-6 loads.

If "A" Charging Pump is also disabled, i.e., all three Charging Pumps disabled, maintaining RCS inventory can be accomplished by use of the High Head Safety Injection Pumps.

9.5.1.3.4.14 Zone 14 - Solid Waste Handling Room

a) ACCESS

Zone 14, the Solid Waste Handling Room is located on the second floor of the Auxiliary Building on the 246 ft elevation. Access to this area is through the Solid Waste Handling Room Door, Fire Door 10, or Security Door 46.

b) COMBUSTIBLES

Combustible material located in Zone 14 consists of Class A material such as clothing and paper. The estimated fire load of this material is less than 40,000 Btu/ft^2 with a maximum fire severity of less than 30 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 14 consists of twelve heat detectors and one manual pull station. These detectors will, through the FDAP-A2 and B2:

1) Actuate the FAP in the Control Room, and

2) Automatically actuate the preaction water deluge valve after sending a signal from at least one detector on train A and one detector on train B.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 14 consists of a preaction water deluge system which is actuated by:

1) Charging the spray header with water

(a) Automatically through the detection system

(b) Manually at the FDAP-A2 and B2, and

(c) Manually at the deluge valve.

2) Melting the fused link at individual spray nozzles.

This system may be isolated by manually closing the isolation valve just below the deluge valve. The deluge valve is located across from the containment. Purge Fan Area in the second floor hallway of the Auxiliary Building in Zone 15.

The following portable fire suppression equipment is available for use in Zone 12:

1) Two 20 lb Dry Chemical Fire Extinguisher (Zone 15), and

2) One Hose Station (Zone 15).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 14 are of noncombustible fire resistant material. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, Fire Door 10 also provides a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 14 contains no safety-related shutdown equipment so a fire in this zone will not inhibit a safe shutdown of the plant.

9.5.1.3.4.15 Zone 15 - Auxiliary Building Second Level Hallway

a) ACCESS

Zone 15, the Auxiliary Building Second Level Hallway is located on the 246 ft elevation of the Auxiliary Building. Access to this area is through Fire Door 14 or Security Door 46.

b) COMBUSTIBLES

Combustible material located in Zone 15 consists of 106 ft^3 cable insulation with a fire load of 44,900 Btu/ft^2. The maximum fire severity for these combustibles located in Zone 15 is 40 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 15 consists of five heat detectors, five ionization smoke detectors, and two manual pull stations. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 15. However, the following portable fire suppression equipment is available:

1) Two 20 lb Dry Chemical Fire Extinguishers (Zone 15), and

2) One Hose Station (Zone 15).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 15 are of noncombustible fire resistant materials. The wall and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, the fire doors in Zone 15 also provide a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 15 does not contain any safety-related shutdown equipment so a fire in this zone does not inhibit safe shutdown of the plant.

9.5.1.3.4.16 Zone 16 - Battery Room

a) ACCESS

Zone 16, the Battery Room, is located on the 248 ft elevation of the Auxiliary Building. Access to this room can be obtained either through Fire Door 11 or Fire Door 12. Fire Door 11 is from the E1-E2 room and Fire Door 12 is from the Batch Tank Room.

b) COMBUSTIBLES

Combustible material located in Zone 16 consists of 13 ft^3 of cable insulation with a fire load of 13,800 Btu/ft^2. In addition, there is liberated hydrogen with a maximum fire severity of 10 min.

c) FIRE DETECTION

Low voltage fire detection in Zone 16 consists of two ionization smoke detectors, two explosion proof heat detectors, and two manual pull stations. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

An additional high voltage system is also available in Zone 16 which consists of one heat detector and one ionization smoke detector which when actuated annunciates on the RTGB in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 16. However, the following portable fire suppression equipment is available:

1) One 20 lb CO_2 Dry Chemical Fire Extinguisher (Zone 16)

2) Two 20 lb CO_2 Fire Extinguishers (Zone 20), and

3) One Hose Station (Zone 20).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 16 are of noncombustible fire resistant materials. The walls and floors are of reinforced concrete and provide a three-hour rated fire barrier. In addition, the fire doors also provide a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 16 contains the station batteries, "A" and "B" battery chargers, and the 125 V DC MCC "A" and "B". A major fire in this zone would result in the loss of station DC power and effectively a loss of maintained 120 V AC control power. This disables plant shutdown control from the Main Control Room.

The loss of DC power also results in solid breakers on the 4160 V and 480 V buses which means all electrically driven plant auxiliaries not required for plant shutdown operations will have to be de-energized by manual local operation of the individual breakers at the buses.

Priority in this case would be to man the remote shutdown monitor stations in the Charging Pump Room and the Turbine Building and to keep the 480 V dedicated shutdown bus energized.

If off-site power is available, the dedicated shutdown bus will be supplied power from the startup transformer via the 4160 V bus 3. If off-site power is not available, the dedicated shutdown diesel generator will supply the bus.

With the startup transformer available and energized, DC control power can be supplied to 4160 V buses 3 and 4 and 480 V buses 2B and 3 via the emergency DC backup system. Transfer to this system is accomplished by manual transfer of safety switches 3 and 4 located in the 4160 V room. Completion of this results in availability of various plant auxiliaries that could be used to assist in plant shutdown although they are not required.

DC control power is not required for the feeder breakers off the dedicated shutdown bus. These breakers utilize AC control power which is supplied off the dedicated bus via a 480/120 V 5kVa transformer.

The dedicated diesel generator output breaker to the bus does require DC control power to trip and this is supplied from a 48 V uninterruptible power supply (UPS) battery.

Therefore, a fire in Zone 16 does have a significant effect on normal shutdown. However, safe shutdown can be accomplished through the use of the dedicated shutdown system.

9.5.1.3.4.17 Zone 17 - HVAC Equipment Room for Control Room

a) ACCESS

Zone 17, HVAC Equipment Room for the Control Room is located on the 242.50 ft elevation of the Auxiliary Building. Access to this area is through either Security Door 41 or 42.

b) COMBUSTIBLES

Combustible material located in Zone 17 consists of 210 lb of charcoal filters. The fire load of this combustible material is 3,600 Btu/ft^2 with a maximum fire severity of 3 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 17 consists of two ionization smoke detectors, two heat detectors, and one manual pull station. These detectors will, through the FDAP A2 and B2, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 17. However, the following fire suppression equipment is available:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 17)

2) One Dry Chemical Fire Extinguisher (North of 4160 Room, next to stairs), and

3) Two 10 lb Dry Chemical Fire Extinguishers (Zone 18).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type,

(i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 17 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, Fire Door 22 which separates Zone 17 and 18 provides a one-and-a-half-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 17 does not contain any safety-related shutdown equipment so a fire in this Zone would not inhibit a safe shutdown of the plant. However, a fire in this Zone might cause a smoke buildup in the Control Room. This would necessitate evacuation of the Control Room and initiation of shutdown operations using the Dedicated Shutdown System.

9.5.1.3.4.18 Zone 18 - Unit No. 1 Cable Spreading Room

a) ACCESS

Zone 18, the Unit No. 1 Cable Spreading Room is located on the 242.50 ft elevation of the Auxiliary Building. Access to this area is through Security Door 43 or through the fire door separating the HVAC Equipment Room for the Control Room and the Unit No. 1 Cable Spreading Room.

b) COMBUSTIBLES

Combustible material located in Zone 18 consists of 36 ft^3 of cable insulation. The fire load of this combustible material is 41,800 Btu/ft^2 with a maximum fire severity of 31 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 18 consists of two ionization detectors and two manual pull stations. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

An additional high voltage system is also available for fire detection in Zone 18 which consists of two ionization detectors which when actuated annunciate on the RTGB in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 18. However, the following portable fire suppression equipment is available:

1) Two 10 lb Dry Chemical Fire Extinguishers (Zone 18)

2) One 20 lb Dry Chemical Fire Extinguisher (Zone 17)

3) One 20 lb CO_2 Fire Extinguisher (Unit 2 Cable Spread Room Entrance)

4) One Hose Station (Unit No. 2 Cable Spread Room Entrance), and

5) One 20 lb Dry Chemical Fire Extinguisher (North of 4160 Room next to stairs.

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 18 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. The fire doors in Zone 18 provide a minimum of one-and-a-half-hour fire rated barrier.

f) SHUTDOWN CAPABILITY

Zone 18 (Unit No. 1 Cable Spreading Room) does not contain any Unit No. 2 safety-related equipment. A fire in this zone will not effect safe shutdown capabilities of Unit No. 2.

9.5.1.3.4.19 Zone 19 - Unit II Cable Spreading Room

a) ACCESS

Zone 19, the Unit II Cable Spreading Room is located on the 242.50 ft elevation of the Auxiliary Building. Access to this room is through Security Door 44 or Fire Door 21.

b) COMBUSTIBLES

Combustible material located in Zone 19 consists of 179 ft^3 of cable insulation with a fire load of 85,100 Btu/ft^2. In addition, there is 24 ft^3 of paper with a fire load of 3,600 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 19 is 66 min.

c) FIRE DETECTION

Low voltage fire detection in Zone 19 consists of two heat detectors, four photoelectric smoke detectors, two ionization smoke detectors, and one manual pull station. These detectors will, through the FDAP-A2 and B2:

1) Actuate the FAP in the Control Room, and

2) Actuate the Halon System with a detection signal from both C/D trains.

An additional high voltage detection system is also available in Zone 19 which consists of two heat detectors and two ionization smoke detectors which when actuated annunciate on the RTGB in the Control Room.

d) FIRE SUPPRESSION

Fixed suppression for Zone 19 consists of a Halon Suppression System. This system is actuated:

1) Automatically by the FDAP which requires a detection signal from both C/D trains.

2) Manually from either FDAP-A2 (at south end of MCC-2) or FDAP-B2 (east of the IVSW Tank Area), by depressing the Zone Suppression Manual Actuation (ZSMA) switches to the "Operated" position.

3) Manually by pulling the safety pins and rotating the pilot control valves HS-8 and HS-11 counterclockwise until they stop. Next, pull safety pins on Manual Pneumatic Actuator Valves HS-18 and HS-20 on bottles A4 and A5 then pull both levers down simultaneously.

This system when actuated automatically or manually from the FDAP provides a 15 sec delay before discharging.

The Halon System can be isolated by rotating the levers to close the ball valves thereby restricting the flow of Halon. Isolation cannot be achieved after activation.

The following portable fire suppression equipment is available for use in Zone 19:

1) One 20 lb CO_2 Fire Extinguisher (Zone 19)

2) Two 20 lb CO_2 Fire Extinguishers (Zone 20)

3) One Hose Station (Zone 19 Entrance), and

4) One Hose Station (Zone 20).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type, (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 19 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. The fire doors in Zone 19 provide a minimum of one-and-a-half-hour fire rated barrier.

f) SHUTDOWN CAPABILITY

Zone 19, Unit 2 Cable Spread Room does contain equipment that is vital to accomplish safe shutdown of the plant.

Equipment and systems that will be disabled on a major fire in this Zone include, Unit 2 Communications, Relay Protection for the Unit 2 (230 kV) Switchyard, Rod Position Detection, EHC Turbine Controls, Main Computer, Auxiliary Relaying, and possibly Unit 2 Annunciation (Alarms).

There will also be a loss of boric acid transfer pump and charging control. A fire in this area could also result in the spurious operation of safety-related valves, breakers, pumps, etc.

A major fire in this area may also require a total evacuation of the Control Room.

In either case, a safe plant shutdown should be accomplished by use of the Dedicated Shutdown System with shutdown monitoring accomplished at the Charging Pump Room Panel and the Secondary Monitor Panel in the Turbine Building.

Priority in the case should be to commence a boration prior to Control Room evacuation and then man the remote panels for a controlled shutdown.

In this case, the Startup Transformer cannot be relied upon for a source of power to the Dedicated Shutdown Bus so the Dedicated Shutdown Diesel should be used as a power supply to the DS Bus.

9.5.1.3.4.20 Zone 20 - Emergency Switchgear Room

a) ACCESS

Zone 20, Emergency Switchgear Room, is located on the 242.50 ft elevation of the Auxiliary Building. Access to this room is through either Fire Door 15 or Security Door 45.

b) COMBUSTIBLES

Combustible material located in Zone 20 consists of 303 ft^3 of cable insulation. The fire load of this combustible material is 70,700 Btu/ft^2 with a maximum fire severity of 53 min.

c) FIRE DETECTION

The low voltage (System 3) fire detection system in Zone 20 consists of six heat detectors, four ionization smoke detectors, four photoelectric smoke detectors, and one manual pull station. These detectors will, through the FDAP-A1 and B1:

1) Actuate the FAP in the Control Room, and

2) Actuate the Halon System with a detection signal from both A/B trains.

An additional high voltage system is also available for fire detection in Zone 20 which consists of six heat detectors and four ionization smoke detectors which when actuated annunciates on the RTGB in the Control Room.

d) FIRE SUPPRESSION

Fire suppression for Zone 20 consists of a Halon Suppression System. This system is actuated:

1) Automatically through the FDAP.

2) Manually from either FDAP-A1 (south of the instrument and service air compressors on the first floor Auxiliary Building Hallway) or FDAP-B1 (across from the Hot Lab) by moving the "Normal/Operated" switch associated with the window marked "Manual Actuation Z-20 Halon" to the "Operated" position.

3) Manually by pulling the safety pins and rotating the pilot control valves HS-2 and HS-5 counterclockwise until they stop. Next pull safety pins on the manual pneumatic actuator valves HS-14 and HS-16 on bottles A1 and A2, then pull both levers down simultaneously.

This system, when actuated automatically or manually from the FDAP, provides a 15 sec delay before discharge.

The Halon System can be isolated by rotating the levers to close the ball valves (HS-4, HS-1) thereby restricting the flow of Halon to Zone 20. This will not restrict flow subsequent to actuation.

The following portable fire suppression equipment is available for use in Zone 20:

1) Two 20 lb CO_2 Fire Extinguishers (Zone 20)

2) One Hose Station (Zone 20), and

3) One Hose Station (Zone 14 Entrance).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 20 are of noncombustible fire resistant material. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, there are seven fire doors which remain closed and fire dampers which automatically close upon detection of a fire.

f) SHUTDOWN CAPABILITY

Safety-related shutdown equipment located in Zone 20 includes emergency buses E1 and E2, MCC 9 and 5, maintained 120 V AC inverters "A", "B", and "C", instrument buses 1, 2, 3, 4, 6, 7A, 7B, 8, 9A and 9B, the reactor protection relay racks, and the safeguards relay racks.

Other equipment located in this zone includes the computer inverter, MCC-2, primary water MOV to the CCW system, service water pumps "A" and "B", remote local selection switches, and containment vent recirculating control selection.

A major fire in this Zone would result in the potential loss of all the previously mentioned equipment and would probably require evacuation of the Control Room. Even if Control Room evacuation was not required, the loss of the E1 and E2 buses, all three DC/AC inverters and all instrument buses would require a controlled plant shutdown to be accomplished from the remote monitor shutdown panels and by use of the Dedicated Shutdown System.

Priority in this case would be to man the remote shutdown monitor panels in the Charging Pump room and the secondary plant and ensure that the 480 V dedicated shutdown bus is energized.

If offsite power is available, the dedicated shutdown bus will be supplied power from the startup transformer via the 4160 V Bus 3. If offsite power is not available, the dedicated shutdown diesel generator will supply the bus.

A fire in this Zone also provides the potential for a loss of all DC control power.

If the startup transformer is available and energized, DC control power to 4 KV buses 3 and 4 and 480 V buses 2B and 3 should be transferred to the emergency backup. Transfer to this system is accomplished by manual transfer of safety switches 3 and 4 located in the 4160 Volt room. Completion of this results in the availability of various plant auxiliary equipments that could be used to assist in plant shutdown although they are not required.

A safe plant shutdown can be accomplished by use of the dedicated shutdown system.

The normal plant 120 V AC system and the 125 V DC system is not required to operate the dedicated shutdown system.

DC control power is not required for the feeder breakers off the shutdown bus. These breakers utilize AC control power which is supplied off the dedicated bus via a 480/120 V 5 kVa transformer.

The output breaker from the dedicated diesel generator does require DC control power to trip and this is supplied from a 48 V UPS battery.

120 V AC instrument power to the Charging Pump room panel and the Turbine Building Remote Shutdown Panel is supplied from the dedicated shutdown system via a battery charger inverter.

A Zone 20 fire would also result in a loss of ability to make up water to the CCW system from the primary water and demineralized water systems but this will not prevent accomplishing a plant shutdown.

9.5.1.3.4.21 Zone 21 - Rod Control Room

a) ACCESS

Zone 21, the Rod Control Room, is located on the second floor of the Auxiliary Building on the 249.50 ft elevation. Access to this area is through either Fire Door 16 or Security Door 53.

b) COMBUSTIBLES

Combustible material located in Zone 21 consists of 44 ft^3 of cable insulation. The fire load of this combustible material is 24,500 Btu/ft^2 with a maximum fire severity of 18 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 21 consists of three ionization detectors and one manual pull station. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

An additional high voltage detection system is also available in Zone 21 which consists of three ionization detectors which when actuated annunciate on the RTGB in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 21. However, the following portable fire suppression equipment is available:

1) One 20 lb CO_2 Fire Extinguisher (Zone 21).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 21 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, the fire door in Zone 21 provides a three-hour fire rated barrier.

f) SHUTDOWN CAPABILITY

Safety-related shutdown equipment in Zone 21 includes the local remote control selection switches for Service Water Pumps "D" and "C". With a fire in this Zone there is the potential for loss of use of these two pumps. This would not inhibit a safe shutdown of the plant with Service Water Pumps "A" and "B" still available for service.

9.5.1.3.4.22 Zone 22 - Control Room

a) ACCESS

Zone 22, the Control Room, is located on the 254 ft elevation of the Auxiliary Building. Access to this area is through either Security Door 48 or Security Door 49.

b) COMBUSTIBLES

Combustible material located in Zone 22 consists of ~18 ft^3 of cable insulation with a fire load of ~5,000 Btu/ft^2. In addition, there is ~240 ft^3 of paper with a fire load rating of ~21,300 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 22 is 16 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 22 consists of four heat detectors and twelve ionization smoke detectors. Six of the twelve ionization smoke detectors are located inside the RTGB. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 22. However, the following portable fire suppression equipment is available:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 22)

2) One 2 1/2 gal Water Fire Extinguisher (Zone 22)

3) Two 20 lb CO_2 Fire Extinguishers (Zone 22)

4) One 2 1/2 lb Dry Chemical Fire Extinguisher (Zone 22), and

5) One Hose Station (Zone 23).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 22 are of noncombustible fire resistant materials. The walls and floor on the entire control level are of reinforced concrete and provide a three-hour rated fire barrier. All fire doors leading into the Control Room are three-hour fire rated barriers.

f) SHUTDOWN CAPABILITY

Zone 22 is the Control Room and a major fire would require evacuation of the Control Room and may precipitate the necessity to perform a plant shutdown from the Remote Shutdown Monitor Panels in the Charging Pump Room and the Turbine Building and by use of the Dedicated Shutdown System.

Priority in this case would be to man the Remote Shutdown Panels and the Dedicated Diesel Generator Remote Control Panel.

If offsite power is available the Dedicated Shutdown Bus will be energized from the startup transformer via 4 kV Bus No. 3. If offsite power is not available the Dedicated Shutdown Diesel Generator will supply power.

With the startup transformer available and energized the DC control power for 4 KV Buses 3 and 4 and 480 V Buses 2B and 3 should be transferred to the emergency backup system. This transfer is accomplished by transferring Safety Switches 3 and 4 located in the 4160 V Room. This should be completed in case the Normal 125 V DC System is lost. This provides reliability of auxiliaries that would assist in a plant shutdown.

If the Dedicated Shutdown Bus is supplied from the Diesel Generator this will provide sufficient shutdown capability without Normal Station DC control power or maintained 120 V AC control power. All necessary control power is supplied by the Dedicated Shutdown System. A safe plant shutdown can be accomplished utilizing this system.

9.5.1.3.4.23 Zone 23 - Hagan Room

a) ACCESS

Zone 23, the Hagan Room is located on the third level of the Auxiliary Building on the 254 ft elevation. Access to this area is through either Fire Door 17 or Fire Door 25.

b) COMBUSTIBLES

Combustible material located in Zone 23 consists of 18 ft^3 of cable insulation with a fire load of 17,000 Btu/ft^2. In addition, there is 22 ft^3 of paper with a fire load of 6,600 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 23 is 18 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 23 consists of two heat detectors, two ionization smoke detectors, and one manual pull station. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 23 consists of a dry standpipe system. This system is actuated manually in the following manner:

1) Removing all Hose off the Reel, and

2) Opening Valve FP-95 at the Hose Connection in the Hagan Room.

This process will subsequently allow free flow of water through the standpipe and hose.

The dry standpipe system can be isolated by closing the valve FP-87.

The following portable fire suppression equipment is available for use in Zone 23:

1) One Hose Station (Zone 23)

2) One 20 lb Dry Chemical Fire Extinguisher (Zone 22)

3) One 2 1/2 gal Water Fire Extinguisher (Zone 22)

4) Two 20 lb CO_2 Fire Extinguishers (Zone 22), and

5) One 2 1/2 lb Dry Chemical Fire Extinguisher (Zone 22).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 23 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. In addition, Fire Door 17 and Fire Door 25 also provide a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 23 contains the incore Nuclear Instrumentation System (NIS) Racks and the control and protection racks. A fire in this Zone would result in the loss of this equipment. This equipment is not vital to a safe shutdown of the plant but a fire in this room would necessitate the evacuation of the Control Room.

A controlled plant shutdown can be accomplished from the Remote Shutdown Monitor Panels in the Charging Pump Room and the Turbine Building and by utilizing the Dedicated Shutdown System.

If offsite power is available the Dedicated Shutdown Bus will be energized from the Startup Transformer via 4KV Bus 3. If offsite power is not available the Dedicated Shutdown Diesel Generator will supply power to the bus.

A fire in this Zone should not result in the loss of normal 125 V DC System on the maintained 120 V AC System. With these systems still available, operation of the plant auxiliaries with local controls can easily be accomplished to support plant shutdown.

9.5.1.3.4.24 Zone 24 - Containment Cable Penetration Area

a) ACCESS

Zone 24, the Containment Cable Penetration Area, is located on the ground level of the Containment Vessel outside the Polar Crane Wall. Access to this area is through the C.V. Personnel Entry, Security Door 30.

b) COMBUSTIBLES

Combustible material located in Zone 24 consists of 133 ft^3 of cable insulation. The fire load of the combustible material is 61,300 Btu/ft^2 with a maximum fire severity of 46 min.

c) FIRE DETECTION

Low voltage fire detection in Zone 24 consists for four ionization smoke detectors, four heat detectors, and one manual pull station. These detectors will, through the FDAP-A2 and B2:

1) Actuate the FAP in the Control Room, and

2) Actuate the preaction water deluge valve after sending a signal from at least one detector from train A and one detector from train B.

d) FIRE SUPPRESSION

Fire suppression for Zone 24 consists of a preaction water deluge system which is activated by:

1) Charging the spray header with water.

(a) Automatically through the detection system

(b) Manually at the FDAP-A2 and B2 or the Containment Fire Protection Panel (Control Room), and

(c) Manually at deluge valve, located in the stairwell to the second level of the Auxiliary Building.

2) Melting the fused link at individual spray nozzles.

This system may be isolated by either:

1) Manually closing the block valve upstream of the deluge valve. The valve is located in the ground floor landing of the stairwell to the second floor of the Auxiliary Building.

2) Remotely closing either of the motor operated isolation valves from the Control Room. These valves are located in the Auxiliary Building Pipe Alley, Zone 2B.

The following portable fire suppression equipment is available for use in Zone 24:

1) Three 20 lb CO_2 Fire Extinguishers (Zone 24), and

2) Four Hose Stations (Containment).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 24 are of noncombustible fire material. The separation of redundant cables in this area and the fire resistance of the silicon-rubber cable jacket afford protection against loss of redundant cabling or spread of a cable fire. Smoke from a fire in this zone will propagate throughout the Containment Vessel.

f) SHUTDOWN CAPABILITY

Zone 24, the Containment Cable Penetration Area, contains cable runs for the Dedicated Shutdown System but all cables associated with this system have been run in separate electrical conduit with the conduit wrapped with fire retardant material. These protective measures prevent the loss of the Dedicated Shutdown System due to a fire in this Zone and thus a safe plant shutdown can be accomplished.

9.5.1.3.4.25 Zone 25A - Reactor Coolant Pump "A"

a) ACCESS

Zone 25A, Reactor Coolant Pump "A" room is located on the ground level of the Containment Vessel inside the Polar Crane Wall. Access to this room is through the C.V. Personnel Entry, Security Door 30.

b) COMBUSTIBLES

Combustible material located in Zone 25A consists of 175 gal of oil in Reactor Coolant Pump "A". The fire load of the combustible material is 18,500 Btu/ft^2 with a maximum fire severity of 14 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 25A consists of one ionization smoke detector and one infrared flame detector. These detectors will, through the FDAP-A2 or B2:

1) Actuate the FAP in the Control Room, and

2) Actuate the preaction water deluge valve after sending a signal from both detectors (train A and train B).

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 25A consists of a preaction water deluge system which is activated by:

1) Charging the spray header with water.

(a) Automatically through the detection system

(b) Manually at the FDAP-A2 and B2 or Containment Fire Protection Panel (Control Room), and

(c) Manually at deluge valve control panel.

2) Melting the fused link at individual spray nozzles.

This system may be isolated by either:

1) Manually closing the block valve upstream of the deluge valve. The valve is located in the Containment Vessel next to the Polar Crane Wall, opposite the Personnel Entry.

2) Remotely closing either of the motor operated isolation valves from the Control Room. These valves are located in the Auxiliary Building Pipe Alley, Zone 28.

3) Manually closing the block valve, upstream of the motor operated isolation valves, which is also located in the Auxiliary Building Pipe Alley, Zone 28.

The following portable fire suppression equipment is available throughout the ground level of the Containment Vessel for use in Zone 25A:

1) Three 20 lb Dry Chemical Fire Extinguishers

2) Three 20 lb CO_2 Fire Extinguishers, and

3) Four Hose Stations.

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 25A are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. Smoke from a fire in this zone will propagate throughout the Containment Vessel.

f) EVALUATION OF SAFE SHUTDOWN

Occurrence of a significant fire within this Zone can result in the loss of the following equipment:

1) Reactor Coolant Pump "A", and

2) All instrumentation for Primary Loop "A".

Loss of this equipment would cause an automatic reactor trip. There would be no impact on the ability to safely shutdown due to the separation and redundancy of the Primary Coolant Loops.

9.5.1.3.4.26 Zone 25B - Reactor Coolant Pump "B"

a) ACCESS

Zone 25B, Reactor Coolant Pump "B" room, is located on the ground level of the Containment Vessel, inside the Polar Crane Wall. Access to this room is through the C.V. Personnel Entry, Security Door 30.

b) COMBUSTIBLES

Combustible material located in Zone 25B consists of 175 gal of oil in Reactor Coolant Pump "B". The fire load of the combustible material is 18,500 Btu/ft^2 with a maximum fire severity of 14 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 25B consists of one ionization smoke detector and one infrared flame detector. These detectors will, through the FDAP-A2 or B2:

1) Actuate the FAP in the Control Room

2) Actuate the preaction water deluge valve after sending a signal from both detectors (train A and train B).

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 25B consists of a preaction water deluge system which is actuated by:

1) Charging the spray header with water.

(a) Automatically through the detection system

(b) Manually at the FDAP-A2 and B2 or the Containment Fire Protection Panel (Control Room), and

(c) Manually at deluge valve control panel.

2) Melting the fused link at individual spray nozzles.

This system may be isolated by either:

1) Manually closing the block valve upstream of the deluge valve. The valve is located in the containment vessel next to the Polar Crane Wall, opposite the personnel entry.

2) Remotely closing either of the motor operated isolation valves from the Control Room. These valves are located in the Auxiliary Building pipe alley, Zone 28.

3) Manually closing the block valve upstream of the motor operated isolation valves which is also located in the Auxiliary Building pipe alley, Zone 28.

The following portable fire suppression equipment is available throughout the ground level of the Containment Vessel for use in Zone 25B:

1) Three 20 lb Dry Chemical Fire Extinguishers

2) Three 20 lb Carbon Dioxide Fire Extinguishers, and

3) Four Hose Stations.

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 25B are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. Smoke from a fire in this zone will propagate throughout the Containment Vessel.

f) EVALUATION OF SAFE SHUTDOWN

Occurrence of a significant fire within this zone can result in the loss of the following equipment:

1) Reactor Coolant Pump "B", and

2) All instrumentation for Primary Loop "B".

Loss of this equipment would cause an automatic reactor trip. There would be no impact on the ability to safely shutdown due to the separation and redundancy of the Primary Coolant Loops.

9.5.1.3.4.27 Zone 25C - Reactor Coolant Pump "C"

a) ACCESS

Zone 25C, Reactor Coolant Pump "C" room, is located on the ground level of the Containment Vessel inside the Polar Crane Wall. Access to this room is through the C.V. Personnel Entry, Security Door 30.

b) COMBUSTIBLES

Combustible material located in Zone 25C consists of 175 gal of oil in Reactor Coolant Pump "C". The fire load of the combustible material is 18,500 Btu/ft^2 with a maximum fire severity of 14 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 25C consists of one ionization smoke detector and one infrared flame detector. These detectors will, through the FDAP-A2 or B2:

1) Actuate the FAP in the Control Room, and

2) Actuate the preaction water deluge valve after sending a signal from both detectors (train A and train B).

d) FIRE SUPPRESSION

Fixed fire suppression for Zone 25C consists of a preaction water deluge system which is actuated by:

1) Charging the spray header with water.

(a) Automatically through the detection system

(b) Manually at the FDAP-A2 and B2 or the Containment Fire Protection Panel (Control Room), and

(c) Manually at deluge valve.

2) Melting the fused link at individual spray nozzles.

This system may be isolated by either:

1) Manually closing the block valve upstream of the deluge valve. The valve is located in the Containment Vessel next to the Polar Crane Wall, opposite the personnel entry.

2) Remotely closing either of the motor operated isolation valves from the Control Room. These valves are located in the Auxiliary Building pipe alley, Zone 28.

3) Manually closing the block valve, upstream of the motor operated isolation valves, which is also located in the Auxiliary Building pipe alley, Zone 28.

The following portable fire suppression equipment is available throughout the ground level of the Containment Vessel for use in Zone 25C:

1) Three 20 lb Dry Chemical Fire Extinguishers

2) Three 20 lb Carbon Dioxide Fire Extinguishers, and

3) Four Hose Stations.

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 25C are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. Smoke from a fire in this zone will propagate throughout the Containment Vessel.

f) EVALUATION OF SAFE SHUTDOWN

Occurrence of a significant fire within this Zone can result in the loss of the following equipment:

1) Reactor Coolant Pump "C", and

2) All instrumentation for Primary Loop "C".

Loss of this equipment would cause an automatic reactor trip. There would be no impact on the ability to safely shutdown due to the separation and redundancy of the Primary Coolant Loops.

9.5.1.3.4.28 Zone 26 - Containment Air Recirculation Units

a) ACCESS

Zone 26, the Containment Air Recirculation Units, are located on the top floor of the Containment Building at the 275 ft elevation. Access to this area is through the C.V. Personnel Entry, Security Door 30 and then up two flights to the third level.

b) COMBUSTIBLES

Combustible material located in Zone 26 consists of 578 ft^3 of cable insulation. The fire load of this combustible material is 53,400 Btu/ft^2.

In addition, there is also 420 lb of charcoal. The fire load of this combustible material is 1,000 Btu/ft^2 with a maximum fire severity of 41 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 26 consists of four photoelectric smoke detectors and four heat detectors. These detectors will, through the FDAP-A2 and B2, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 26. However, the following portable fire suppression equipment is available:

1) Two 20 lb Dry Chemical Fire Extinguishers (Zone 26), and

2) Four Hose Stations (located in Containment).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 26 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier. Smoke from a fire in this Zone will propagate throughout the Containment Vessel.

f) SHUTDOWN CAPABILITY

Zone 26 contains the Containment Air Recirculation Units and these are not vital to a safe plant shutdown during a fire situation. A fire in this Zone will not inhibit a normal plant shutdown.

9.5.1.3.4.29 Zone 27 - RHR Pit

a) ACCESS

Zone 27, the RHR Pit is located on the 203 ft elevation of the Auxiliary Building. Access to the RHR Pit area is through the Spent Fuel Pool Heat Exchanger Room, Security Door 37 and access to the RHR Pit is through the RHR Pit Hatch, Security Door 29.

b) COMBUSTIBLES

Combustible material located in Zone 27 consists of sixteen gallons of oil. The fire load of this combustible material is 6,200 Btu/ft^2 with a maximum fire severity of five minutes.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 27 consists of two ionization smoke detectors, two heat detectors, one photoelectric smoke detector, and one manual pull station. These detectors will, through the FDAP-A2 and B2 actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 27. However, the following portable fire suppression equipment is available:

1) One 20 lb Dry Chemical Fire Extinguisher (Zone 27)

2) One Fire Hose Reel w/100 ft Hose (Zone 27), and

3) One 20 lb Dry Chemical Fire Extinguisher (Zone 8).

NOTE: All fire suppression equipment in Zone 27, the RHR Pit is located on the level of the entrance hatch, to the RHR Pit, Security Door 29. No fire suppression equipment is located in the RHR Pit.

Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 27 are of noncombustible fire resistant materials. The walls, floor, and hatch are of reinforced concrete and provide a three-hour rated fire barrier.

f) SHUTDOWN CAPABILITY

Zone 27 contains the 2 Residual Heat Removal Pumps. The pumps are in separate bays so a fire in one bay should not affect the other pump. The RHR pumps are required to achieve cold shutdown of the plant. However, a fire in this zone will not inhibit the ability to safely shutdown the plant to a hot shutdown condition.

9.5.1.3.4.30 Zone 28 - Pipe Alley

a) ACCESS

Zone 28, the Pipe Alley, is located on the ground floor of the Auxiliary Building on the 226 ft elevation. Access to this area is off of Zone 11, left of the condensate tanks, through a locked radiation door.

b) COMBUSTIBLES

Combustible material located in Zone 28 consists of 130 ft^3 of cable insulation with a fire load of 30,500 Btu/ft^2. In addition, there is 184 ft^3 of stored clothing with a fire load of 24,300 Btu/ft^2. The maximum fire severity for both combustibles located in Zone 28 is 44 min.

c) FIRE DETECTION

Low voltage (System 3) fire detection in Zone 28 consists of five ionization smoke detectors, two heat detectors, and one manual pull station. These detectors will, through the FDAP-A1 and B1, actuate the FAP in the Control Room.

d) FIRE SUPPRESSION

There is no fixed fire suppression system available in Zone 28. However, the following portable fire suppression equipment is available:

1) One Hose Station (Zone 28)

2) One 20 lb CO_2 Fire Extinguisher (Zone 28)

3) One 20 lb CO_2 Fire Extinguisher (Zone 12), and

4) One Hose Station (Zone 12).

NOTE: Periodically, the recharging or repair of a fire extinguisher may necessitate the replacement of one extinguisher with a different type (i.e., Carbon Dioxide to Dry Chemical). However, all substitutions will be equal to the type and size of the extinguisher it is replacing. All substitutions must be approved by the Fire Protection Senior Specialist.

e) FIRE PROPAGATION CONTROL

The structural features of Zone 28 are of noncombustible fire resistant materials. The walls and floor are of reinforced concrete and provide a three-hour rated fire barrier which isolate Zone 28 from the nearest safety-related area.

f) SHUTDOWN CAPABILITY

Zone 28 (Pipe Alley) does contain safety-related cables for shutdown equipment. A major fire in this zone would result in potential loss of ability to operate various MOV required for Containment Isolation and would render inoperable one train of RHR System related equipment. System redundancy and the fail-safe feature of containment isolation valves would preclude a fire in this area from adversely impacting safe shutdown.

9.5.1.4 Inspection and Testing Requirements

In order to verify the ability of the Fire Protection System components to function as required, a periodic testing and surveillance program was implemented and is given in the Technical Specifications.

9.5.1.5 Organization and Responsibilities

The Technical Specifications, (Section 6.0) for HBR 2, describe the offsite and onsite organization. Within this organization, specific responsibilities are assigned for the Fire Protection Program as follows:

a) Fire Protection Senior Specialist

The Fire Protection Senior Specialist reports to the Operations Supervisor and is responsible for:

1) Coordination of all fire protection program activities

2) Preparation of procedures and instructions which implement the Fire Protection Program

3) Ensuring the development and technical adequacy of the training materials and training sources related to the fire protection program, and assigning qualified Fire Protection instructors

4) Preparation of the listing of those Fire Protection items which are subject to the quality assurance procedures delineated in Section 9.5.1.10

5) Periodic monitoring of all fire protection activities

6) Assisting the Plant Supervisors in assuring that all corrective maintenance and modifications of the fire protection system comply with the Technical Specifications

7) Coordination of the arrangements for offsite fire company support and training as delineated in Sections 9.5.1.9 and 9.5.1.7.2

8) Scheduling and implementation of the Fire Drills Program in accordance with Section 9.5.1.8

9) Establishing the minimum equipment for the Fire Brigade Teams in accordance with Section 9.5.1.6.6, and

10) Being a certified Fire Brigade member.

b) Fire Protection Specialist

The Fire Protection Specialist is responsible for:

1) Advising and assisting the Fire Protection Senior Specialist on any fire protection items

2) Maintaining plant operating manuals and procedures current for fire protection

3) Ensuring the Fire Protection Program on Units 1 and 2 is current and effective

4) Performing fire protection technical reviews of plant modification and procedure changes

5) Assisting Senior Specialist in responses to correspondence relating to fire protection

6) Ensuring periodic tests are technically correct. Upgrading periodic tests when required by system modification or changing regulations

7) Preparing procedures and system descriptions for Fire Protection system and equipment

8) Being a certified fire brigade member

9) Planning and conducting initial and periodic training for Fire Protection Technical Aides and the Fire Brigade personnel in fire fighting, fire protection systems, procedures, and equipment, and

10) Planning and conducting training sessions for plant and contracting personnel concerning the fire protection topics.

c) Fire Protection Technical Aide

The Fire Protection Technical Aide is responsible for:

1) Performing routine fire inspections of Units 1 and 2 to ensure compliance with the HBR Fire Protection Manual, and documenting results

2) Conducting and documenting periodic inspections, tests, and preventive minor maintenance of fire protection systems and equipment to ensure proper operational condition

3) Conducting training for fire brigade personnel, plant employees and contracting personnel in Fire Protection systems, equipment, procedures, and other related topics

4) Being certified as a Fire Brigade member. During a fire emergency will function as an advisor to the Fire Brigade Team Leader and can function in any capacity on the Fire Brigade as directed by the Team Leader

5) Supervising and following-up all valve closures or impairments to any fire protection systems or equipment to ensure adequate back-up protection is provided as required by Technical Specifications and to prevent extended or unnecessary impairments

6) Functioning as advisor to Units 1 and 2 Shift Foreman concerning fire protection matters

7) Preparing Fire Reports in accordance with HBR Fire Protection Procedure

8) Assisting the Fire Brigade Team Leader with completing the listing of the on shift fire brigade team members, and

9) Assisting the Fire Protection Senior Specialist or Specialist on any fire protection items or other assigned projects or task.

d) Shift Foreman

The on-duty Shift Foreman is responsible for:

1) Operation of the fire detection and fire suppression systems in accordance with the Technical Specifications and the established operating procedures.

2) Ensuring the availability of at least five shift fire brigade members in accordance with Technical Specification (Section 6.2.2.f).

3) Providing general direction and support to the Fire Brigade Team Leader in the event of a fire. (If the Emergency Coordinator is activated in accordance with Volume 13 of the Plant Operating Manual, this general guidance and support may be provided by the Emergency Coordinator in lieu of the Shift Foreman).

4) Assessing potential and actual impact of fire emergencies on plant operation and altering the plant operations as necessary in accordance with the established procedures.

5) Ensuring the assignment of Fire Watch Personnel in accordance with Section 9.5.1.6.5.

e) Fire Brigade Team Leader

The Fire Brigade Team Leader is responsible for:

1) Direct supervision of the Fire Brigade Team during fire emergencies and fire drills in accordance with Section 9.5.1.6.2 and Fire Protection Procedure FPP-001, "Fire Emergency"

2) Keeping the Shift Foreman (or Emergency Coordinator) appropriately informed on the status of the fire emergency or drill

3) Coordinating the fire fighting activities of offsite fire company support, and

4) Assisting the Fire Protection Technical Aide in completing the on shift fire brigade form.

f) Fire Brigade Team Members

The Fire Brigade team members are responsible for fire fighting in accordance with Section 9.5.1.6.2 and FPP-001, "Fire Emergency", under the direction of the Fire Brigade Team Leader.

g) Security Force Personnel

Security Force personnel are responsible for providing support to offsite fire company personnel in accordance with Section 9.5.1.9 and FPP-001, "Fire Emergency".

9.5.1.6 General Fire Protection Operations Information

9.5.1.6.1 Operation, Maintenance, and Testing

Fire detection systems and fire suppression systems will be operated in accordance with applicable System Operating Procedures. Surveillance testing of these systems will be performed in accordance with the applicable Technical Specifications. Shift Foremen will ensure the fire detection and suppression systems are operated in compliance with the applicable Technical Specifications and System Operating Procedures.

9.5.1.6.2 Fire Emergency Operations

The fire emergency operations are:

a) Any person discovering a fire is responsible for promptly notifying the Unit 2 Control Room Operator upon discovery of a fire

b) Upon receipt of notice of a fire or annunciation of a fire alarm, the Shift Foreman will implement FPP-001, "Fire Emergency"

c) Upon annunciation of a fire alarm or when directed by the Shift Foreman, the Shift Fire Brigade will respond with appropriate fire fighting and protective equipment, and

d) After the fire is extinguished, the Fire Protection Technical Aide will initiate a Fire Report.

9.5.1.6.3 Control of Combustible Materials

Only those storage areas designated in accordance with the Fire Protection Housekeeping Procedure, will be used for storage of materials, equipment, and supplies. Storage outside designated areas is prohibited in zones and areas containing safety-related equipment. Transient combustibles required by maintenance or modifications activities in zones and areas containing safety-related equipment will be controlled in accordance with Fire Protection Procedures. Flammable liquids and gases will be handled in accordance with approved procedures. Area cleanliness will be maintained in accordance with Fire Protection Housekeeping Procedures.

9.5.1.6.4 Control of Ignition Sources

Whenever maintenance or modification involves "Hot Work", a Hot Work Permit and a Fire Watch will be required in accordance with Fire Protection Procedures.

9.5.1.6.5 Fire Watches

Fire watches will be assigned as required by Technical Specifications and whenever maintenance or modification involves "Hot Work". The Shift Foreman will ensure the assignment of fire watches. The cognizant Maintenance Foreman or Construction Representative will ensure the assignment of fire watches for maintenance and modification activities.

Prior to assignment as a fire watch, the individual will satisfy the training requirements for fire watch persons as described by Section 9.5.1.7. Individual fire watch duties will be performed in accordance with Fire Protection Procedures.

9.5.1.6.6 Fire Brigade Teams

The Plant General Manager will designate in writing those individuals who will serve as Fire Brigade Team Leaders and Fire Brigade Team Members. For each shift, the Fire Brigade Team will consist of a Team Leader (who will be the most senior Reactor Operator not assigned to the RTGB), two qualified Auxiliary Operators, the Unit 1 Shift Foreman, and one Unit 1 Auxiliary Operator. An equivalently trained person may be substituted for any fire brigade member. The Fire Brigade members' qualifications will include an annual physical examination for performing strenuous firefighting activity. The Unit 2 Foreman will not be a member of the fire brigade.

The Fire Protection Senior Specialist will establish the minimum satisfactory equipment to be used by the fire brigade teams in accordance with Fire Protection Procedures. The Fire Protection Senior Specialist is also responsible for ensuring this designated Fire Fighting Equipment is obtained, inspected, stored properly, and that the control of this equipment is documented in accordance with Section 9.5.1.10. The offsite fire companies will provide their own protective respiratory equipment.

9.5.1.7 Training

The training requirements for all levels of individuals involved in the fire protection program are described below. This training, and the drills program described in Section 9.5.1.8, provide the means for both ensuring and

evaluating the competence of the fire protection program and the personnel involved in its implementation.

9.5.1.7.1 Training Responsibilities

The Plant General Manager has the overall responsibility for the Fire Protection System Training Program. Specific training responsibilities are delineated as follows:

a) The Training Supervisor is responsible for supporting the fire protection program training by:

1) Assistance in scheduling all training required to fulfill the fire protection courses

2) Assisting in the development of training materials and sources, and

3) Maintaining the fire protection program training records in accordance with Section 9.5.1.7.3.

b) The Fire Protection Senior Specialist is responsible for:

1) Ensuring the development and technical adequacy of the training materials and training sources to be used for the fire protection training program.

2) Periodic surveillance of the fire protection training and documentation to ensure that it is accomplishing the purpose of the fire protection program.

3) Nominate, for approval by the Plant General Manager, those subunit supervisors, engineers, and specialists who will be required to complete the Fire Prevention and Fire Emergency Course.

4) Coordination with offsite fire companies to schedule and accomplish the Offsite Fire Company Training program.

5) Selection and assignment of qualified fire protection instructors.

c) The Plant General Manager will designate, from those personnel nominated by the Fire Protection Specialist, the subunit supervisors, engineers, and specialists who will be required to complete the Fire Prevention and Fire Emergency Course.

d) The Maintenance Supervisors, Maintenance Foremen, and Construction Representative are responsible for ensuring that a sufficient number of persons have received fire watch training to meet the need for fire watches by operations, maintenance, and modifications activities.

e) All Subunit Supervisors will advise the Fire Protection Senior Specialist of those persons under their cognizance who require fire protection training and upon request will provide necessary information on plant activities to permit development of optimum training schedules.

9.5.1.7.2 Training Program

The fire protection training program is designed to provide training to all station personnel commensurate with their respective responsibilities. The following nine training courses are based on the various fire protection responsibilities of station personnel:

a) The General Orientation in Fire Protection Course is required for all persons granted unescorted access. Such persons will complete this course upon initial employment, and annually thereafter. This course may be given integrally with General Orientation Training for station personnel.

b) The Fire Command Course is required for Fire Brigade Team Leaders before assignment.

c) The Fire Fighting Course is required for Fire Brigade Team Leaders and Members before assignment.

In addition, quarterly training sessions will be held such that reviews, updates, or advanced training on the major topics of the Fire Fighting Course are repeated every two years. These quarterly training sessions may be integral with the fire drills.

d) The Fire Detection and Suppression Systems Course will be completed by candidates for Senior Reactor Operator, Reactor Operator, and Auxiliary Operator. This course may be integral with systems training programs for licensed and non-licensed operations personnel.

e) The handling of Flammable or Toxic Liquids, Gases, and Chemicals Course will be completed by Maintenance Foremen, Environmental and Radiation Control Foremen, Stores Foremen, and any other appropriate personnel designated by them. This training should be completed prior to handling such materials and repeated at two year intervals.

f) The Fire Prevention and Fire Emergency Course will be completed upon initial designation and repeated at two year intervals. Subunit supervisors, engineers, and specialists designated by the Plant General Manager will complete this course.

g) Fire Protection Program Management Courses will be completed by the Fire Protection Senior Specialist, Technical Aide, and Specialist such that at least one seminar or course of instruction is completed annually.

h) The duties of Fire Watch Course will be completed by fire watch persons prior to initial assignment to a Fire Watch and repeated at two year intervals.

i) The indoctrination of Offsite Fire Companies Course will be completed annually. Those offsite fire companies which could be called for assistance in a plant fire emergency will be urged to attend this course.

9.5.1.7.3 Records of Training

The Training Supervisor is responsible for documenting and maintaining all Fire Protection Program training in accordance with Fire Protection Procedures. The Fire Protection Senior Specialist will annually audit the Fire Protection

Program training records to ensure that all program training goals are being achieved.

9.5.1.8 Fire Drills Program

The Fire Protection Senior Specialist is responsible for both the scheduling and implementation of the fire drills program. The following factors will be considered when preparing a drill schedule.

a) The fire drill schedule must be coordinated with the Emergency Planning Coordinator and Operations Supervisor

b) Fire drills may be integral with other Plant Emergency drills, and

c) Fire drills must be conducted often enough to fulfill the fire drill frequency requirements below.

Fire drills will be conducted at a frequency which accomplishes the following:

a) Each shift fire brigade team will be drilled at least once per calendar quarter

b) At least one fire drill per year for each shift fire brigade team will be unannounced

c) At least one fire drill per year for each shift fire brigade team will be conducted on other than the day shift, and

d) At least one fire drill per year will include participation by the offsite fire company.

Each fire brigade team member should participate in every drill for that team, and must participate in at least two drills per year.

The Fire Protection Senior Specialist should be guided by the following considerations when planning and critiquing drills:

a) Fire drills should be performed at the plant. The selected situation should simulate the size and arrangement of a fire which could reasonably be expected to occur in the selected area, allow for fire development (due to the time required for personnel to respond, obtain equipment, organize), and sometimes assume that any installed automatic fire suppression system fails to function.

b) The area and type of fire chosen for drills should be varied such that the brigade members are trained in fighting fires in a variety of safety-related and high hazard areas.

c) All unannounced drills may be planned and critiqued in association with the Fire Protection Senior Specialist, the appropriate Operations Supervisor for Unit 2, and the Security Specialist, offsite fire department or their designees. These individuals will ensure that the applicable fire brigade team members are unaware of the drill until it has begun.

d) At 3-year intervals, a fire drill will be critiqued by a qualified outside individual. A copy of this individual's written report will be submitted to the NRC. (This critique may be combined with the triennial audit requirements of Technical Specifications.)

e) Each drill critique should access the following items as appropriate for the planned drill:

1) Fire alarm effectiveness

2) Time required for Fire Brigade Team to respond

3) The selection, placement, and use of equipment

4) Firefighting strategy used

5) The Fire Brigade Team member's knowledge of their roles and conformance with the FPP-001, "Fire Emergency"

6) The use of firefighting equipment, protective respiratory equipment, communication equipment, and ventilation equipment, and

7) The thoroughness, accuracy, and effectiveness of the Fire Brigade Team Leader's firefighting supervision.

The Fire Protection Senior Specialist will obtain, document, and ensure implementation of required corrective actions based on the drill critique recommendations. Any performance deficiencies of a Fire Brigade Team or its members will be remedied by scheduling additional appropriate training. If the overall drill response is judged to be unsatisfactory by the observers, than a repeat drill must be scheduled within 30 days.

The Fire Protection Senior Specialist is responsible for documenting and maintaining the Fire Drills Program records in accordance with procedures.

9.5.1.9 Offsite Fire Company Support

The purpose of offsite fire company support is to supplement the plant's fire fighting capability. To effectively utilize this support, indoctrination training and fire fighting coordination will be conducted as discussed below.

The Fire Protection Senior Specialist is responsible for establishing and maintaining support agreements with the appropriate offsite fire companies. This support should be integrated into both the fire protection training and drills programs. The Fire Protection Senior Specialist will coordinate the training of involved offsite fire companies.

Requests for support by offsite fire companies and coordination of that support will be in accordance with FPP-001, "Fire Emergency".

During a declared emergency, whether or not a drill, offsite fire company persons, materials, and vehicles will be admitted to the protected area without search or log in at entrance. The dosimetry devices and thermoluminescent dosimeter (TLD) provided by the Radiation Control Supervisor will be issued to each fire company person who enters the protected area. The Security Force will

escort responding offsite fire companies within the protected area as required by the plant Security Plan. Prior to permitting offsite fire companies to leave the protected area, radiation control (RC) personnel will log each offsite fire company person, their dosimetry identification number, each vehicle identification number, collect the dosimetry devices and TLD, and complete the appropriate documentation.

9.5.1.10 Quality Assurance Program

The Plant General Manager will direct a documented program of quality assurance for all items designated by the Fire Protection Senior Specialist as Fire Protection items. The program will accomplish the following:

a) Quality Control inspection of the installation, corrective maintenance, modifications, and receipt of all designated fire protection items

b) Verification of compliance with governing procedures of the Fire Protection Program, and

c) Provision for adequate quality assurance controls for all designated fire protection items to ensure the maintenance of an effective fire protection program.

9.5.1.10.1 Design and Modification Control And Documentation

All design changes and modifications to fire protection items will be prepared, reviewed, approved, accomplished, and documented in accordance with the approved Plant Operating Procedures for design changes and modifications.

9.5.1.10.2 Control of Purchased Materials, Equipment, and Services

Control of purchased materials, equipment, and services with respect to fire protection items will be accomplished in accordance with the following specific guidelines for Procurement, Receiving, and Storage.

a) Procurement

1) Purchase Requisitions for fire protection items will be completed in accordance with approved Plant Operating Procedures for non-Q list items, and

2) Purchase Orders for fire protection items will be completed in accordance with approved Plant Operating Procedures for non-Q list items.

b) Receiving

1) All fire protection items will be receipt inspected by the storekeeper in accordance with approved Plant Operating Procedures

2) A Receipt Inspection Report will be completed, and any nonconformities will be documented in accordance with approved Plant Operating Procedures, and

3) All fire protection items will be tagged in accordance with approved Plant Operating Procedures.

c) Storage

All fire protection items will be packaged for storage and will be stored in accordance with approved Plant Operating Procedures.

9.5.1.10.3 Deficiencies and Nonconformance Items

Deficiencies and nonconformances of fire protection system items will be identified, reported, dispositioned, and corrected in accordance with approved Plant Operating Procedures.

9.5.1.10.4 Quality Control Inspections

A documented program of quality control inspections will be implemented which will accomplish the following:

a) Inspection of corrective maintenance and modification activities which add, replace, or repair any of the fire penetration seals, fire retardant coatings, fire barriers, fire dampers, emergency lighting, and communications equipment. This inspection will be accomplished in accordance with Fire Protection Procedures.

b) Inspection of corrective maintenance and modification of fire protection systems which include any of the attributes, adjustment of setpoints, reclosure of fire water suppression systems, and post-maintenance or post-modification testing.

c) Inspection of any installation of fire detectors, fire pumps, and automatically-actuated fire water valves.

9.5.1.10.5 Periodic Surveillance of Compliance with Procedures

The General Plant Manager is responsible for implementing a documented program of periodic surveillance which verifies compliance with governing procedures for the following fire protection activities:

a) Housekeeping

b) Surveillance Tests of the fire protection systems

c) Control of ignition sources

d) Use of fire watches

e) Control of combustibles

f) Fire protection training documentation, and

g) Preventive Maintenance Program.

This program will be conducted by the Fire Protection Senior Specialist in accordance with Fire Protection Procedures.

9.5.1.10.6 Preventive Maintenance

A preventive maintenance program for all designated fire protection system items is the responsibility of the Maintenance Supervisor and will be established and implemented in accordance with approved Plant Operating Procedures for preventive maintenance.

9.5.1.10.7 Post-Maintenance and Post-Modification Testing

Corrective maintenance which repairs or replaces parts or components which affect the function of designated fire protection items requires post-maintenance testing.

Each modification of designated fire protection items requires testing to demonstrate that design criteria and the function of the modification are met. The specific post-maintenance and post-modification test requirements will be delineated in accordance with approved Plant Operating Procedures for post-maintenance and post-modification testing.

9.5.1.10.8 Audits

The Manager, Corporate Quality Assurance Department is responsible for a bi-annual audit in accordance with Technical Specifications which assess the effectiveness of the Fire Protection Program.

Corporate Nuclear Safety and Research is responsible for an independent review of any unreviewed safety questions in accordance with Technical Specifications.

The Fire Protection Senior Specialist is responsible for ensuring the following audits are accomplished:

a) An annual independent audit by qualified offsite personnel or an outside firm in accordance with Technical Specifications, and

b) A triennial audit by an outside fire consultant in accordance with Technical Specifications.

9.5.1.10.9 Quality Assurance Records

Quality Assurance records generated by implementation of this Fire Protection System Quality Assurance Program will be maintained in accordance with Plant Operating Manual, Administrative Instructions.

9.5.1.10.10 Contractual Services Requiring Fire Watches

The Supervisor of any subunit initiating a contract, authorizing letter or purchase requisition for onsite contractor services which include fire watch services must ensure that all fire watches attend the Fire Watch Personnel Training Course or provide written evidence to the satisfaction of the Fire Protection Senior Specialist that an equivalent course has been completed. In addition, contract fire watches must perform their duties in accordance with Fire Protection Procedures.

TABLE 9.5.1-1

FIRE DETECTOR LOCATIONS

Zone 1 - Diesel Generator B

The ultraviolet (UV) Detectors are mounted at a height of 10 ft. This is highest possible height due to pipe obstructions. One detector is mounted on each side of the diesel generator (DG) for optimum protection. Sensitivity is field adjustable for maximum performance.

Zone 2 - Diesel Generator A

Same as Zone 1

Zone 3 - SIS Pump

Cable tray exist on both sides. Detectors mounted as close as possible to cable trays. Beam depth 12 in. Detector types cross-zoned for maximum coverage.

Zone 4 - Charging Pump

Beam depth 12 in. Detector types cross-zoned for maximum coverage.

Zone 5 - Component Cooling

Beam depth 12 in. Detector type cross-zoned for maximum performance.

Zone 6 - Hot Chem. & Count Room

Lay-in ceiling tile. Detectors located using engineering judgment.

Zone 7 - Aux. feedwater (FW) Pump

Same as Zone 6.

Zone 8 - Boron Injection

Beam depth 12 in. Detectors located using engineering judgment.

Zone 9 - Cable Vault N

Beam depth 12 in. Detector types cross-zoned. Detectors located over cable trays. Air handler discharge grilles located at 10 ft level, well below detector mounting heights.

Zone 10 - Cable Vault S

Detector types cross-zoned. Detectors located over cable trays. Photoelectric detectors located for optimum back up efficiency. Beam depths 12 in.

TABLE 9.5.1-1 (Cont'd)

Zone 20 - Electrical Equipment

Beam depth 12 in. Detector types installed alternately on fire circuit A. B circuit detectors installed cross-zoned with circuit A. Detectors installed primarily over cable trays and motor control centers.

Zone 21 - CRDM

Beam depth 12 in. Detectors located over cable trays.

Zone 22 - Control Room

Lay-in ceiling tile. Detectors alternately installed on each fire circuit. Each adjacent detector circuit type is alternately installed. Detectors are primarily located over control cabinets.

Zone 23 - Relay Room

Lay-in ceiling tile detector types and circuits are cross-zoned for optimum coverage. Detectors primarily spaced over relay racks.

Zone 24 - Reactor Building - Elevation 228 ft

Detectors are mounted to the underside of the catwalk grating with primary coverage over the cable trays. Detectors are installed alternately by type, on each fire circuit to provide optimum coverage. Adjacent detectors will also be installed alternately.

Zone 25 - Primary Coolant Pumps

Detectors are installed primarily for the protection of the coolant pumps and detection of fire resulting from spilled oil. Detectors are installed adjacent to each pump motor.

Zone 26 - Containment General Area

Heat Detectors are mounted under the motor housing. Ionization detectors are mounted on the underside of the large duct as close as possible to charcoal filters in the return air supply.

Zone 27 - RHR Pump Pit

Detectors installed over RHR pumps. Combustion of oil is primary fire hazard.

Zone 28 - Pipe Alley

Detectors primarily located in center of hallway. Detector types alternately installed on each circuit. Each circuit alternately interlaced with each other to provide optimum coverage by detector type.

9.5.3 LIGHTING SYSTEMS

In addition to the normal plant lighting, emergency lighting is provided by the station DC power. Because the plant lighting systems are divided into a number of circuits, a fire in an area could cause loss of both normal and emergency lighting in the fire area, but would not cause loss of lighting to areas served by other circuits.

A number of battery-operated portable lanterns are stored in the Control Room and the Auxiliary Building; these lanterns are dedicated for emergency use and are sufficient in number for the fire brigade. Also, battery-powered lights are installed in key areas.

REFERENCES: SECTION 9.5

9.5.1-1 Letter, NO-80-896, dated June 12, 1980, Item 3.2.7, Fire Water Pipe Rupture, from CP&L to NRC.

9.5.1-2 Davis, C.V. and Sorensen, K.E. Eds., Handbook of Applied Hydraulics, (Third Edition), McGraw-Hill Book Co., New York (1969), pp 2-10 and 37-41.

CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM

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STEAM AND POWER CONVERSION SYSTEM

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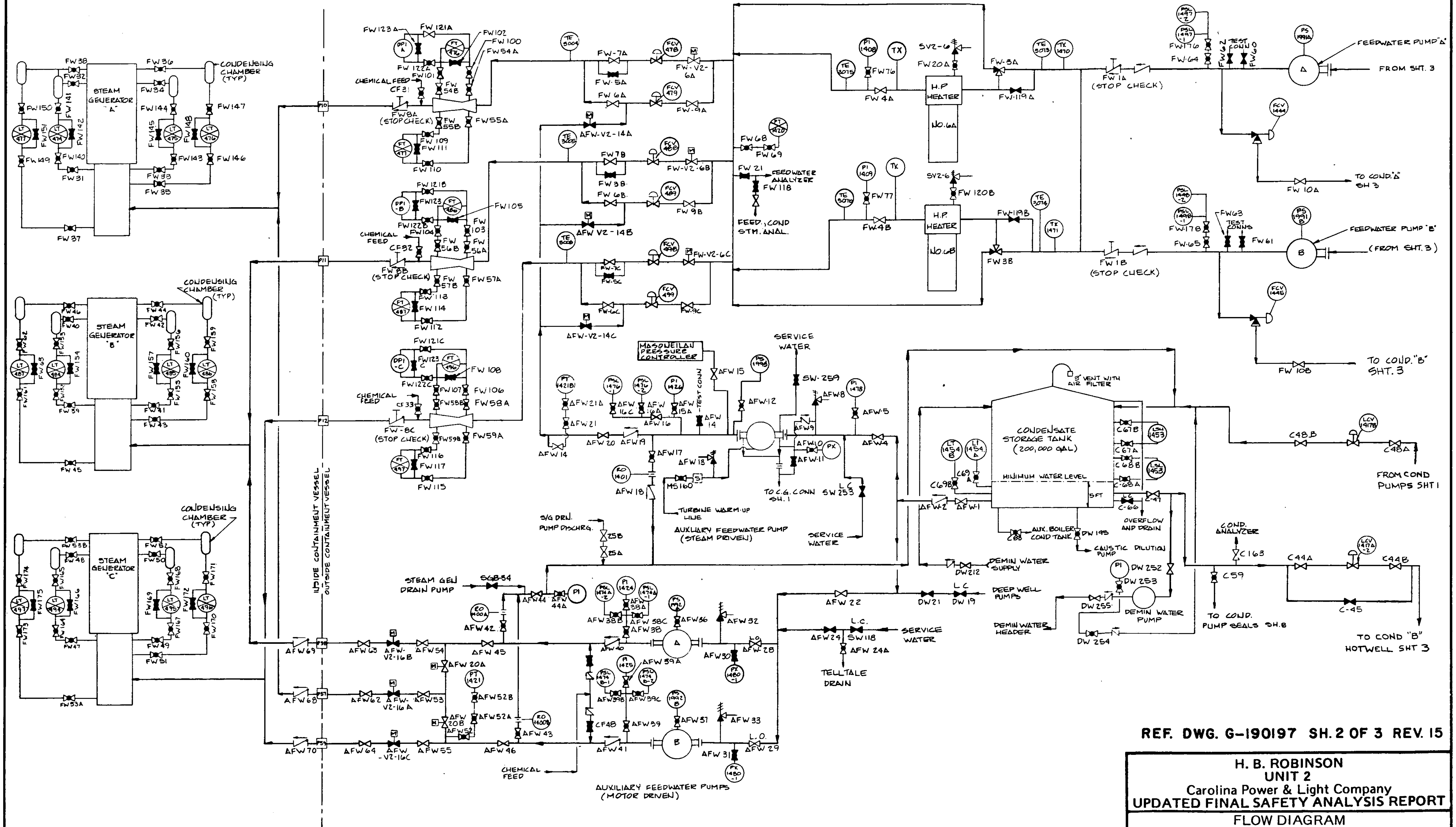
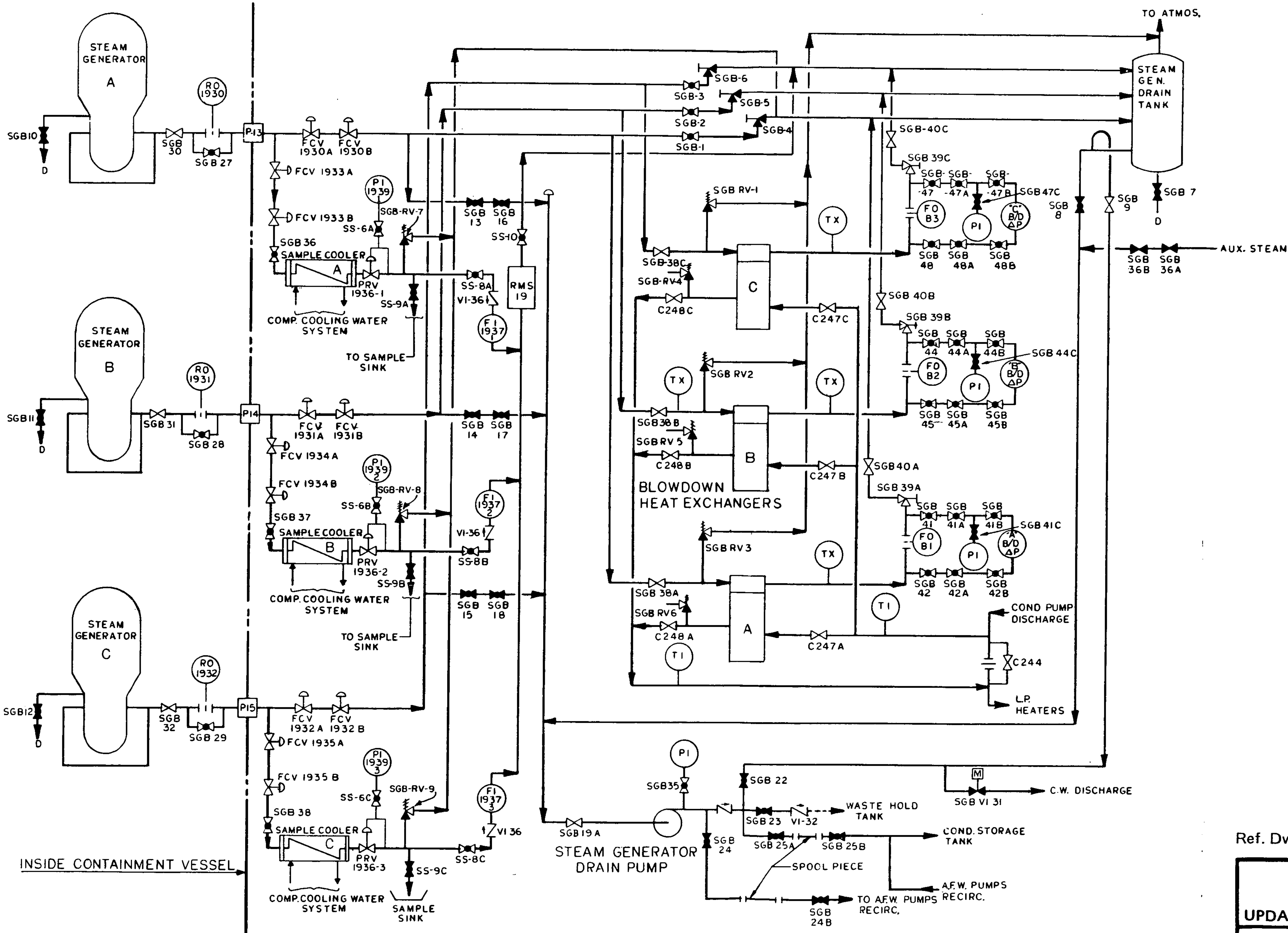


REF. DWG. G-190197 SH. 2 OF 3 REV. 15

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM
FEEDWATER, CONDENSATE, AND AIR EVACUATION SYSTEM
SHEET 2 OF 3
FIGURE 10.1.0 - 5



Ref. Dwg. G-190234 Sheet 1 Rev. 10

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM
STEAM GENERATOR
BLOW DOWN SYSTEM
FIGURE 10.1.0 - 8

10.3.2 SYSTEM DESCRIPTION

The main steam supply system is shown on Figures 10.1.0-1 through 10.1.0-3. Steam from each of the three steam generators is supplied to the turbine, where the steam expands through the high pressure (HP) turbine, and then flows through reheaters and intercept valves to two, double-flow, low-pressure (LP) turbines, all in tandem. Six stages of extraction are provided, two from the HP turbine, one of which is the exhaust, and four stages from each of the LP turbines. The feedwater heaters for the lowest two stages are located in the condenser neck. All feedwater heaters are horizontal, half-size units (two strings). The feedwater string is the closed type with deaeration accomplished in the condenser hotwell to less than 0.005 cc per liter of residual oxygen.

There are four, horizontal-axis, cylindrical-shell, combined moisture-separator, live-steam reheater assemblies. Steam from the exhaust of the HP turbine element enters each assembly at one end. Internal manifolds in the lower section distribute the wet steam. The steam then rises through a wire mesh moisture separator where the moisture is removed. Live steam from the steam generators enters at the other end of each assembly, passes through the tubes and leaves as condensate. The lower pressure steam leaving the wire mesh separator flows over the tube bundle where it is reheated. This reheated steam leaves through openings in the top of the assemblies and flows through individual stop and intercept valves to the LP turbines.

10.3.2.1 Main Steam Isolation Valves

Steam from each of the steam generators flows through a 26 in. swing disc type isolation valve and a swing disc check valve to a 72 in. common header. The main steam line isolation valve bodies are cast carbon steel with stainless steel trim. The valves each pass 3,356,282 lb of steam/hr at 769 psig and 513°F with 0.25 percent moisture. The design pressure and temperature of the valves are 1085 psig and 600°F, respectively. The isolation valves are equipped with three top mounted air cylinders and stay in the open position in the event of loss of air pressure. A bypass valve is provided around each isolation valve to equalize pressure across the valve and for steamline warmup.

10.3.2.2 Main Steam Safety Valves

The steam generator safety valves provide emergency pressure relief for the steam generators as a result of imbalance between steam generation and steam consumption. These valves have a total flow capacity equal to steam generator flow at maximum calculated plant operation at the pressure setting established by the applicable ASME code.

There are four safety valves located on each of the three 26 in. main steam lines outside the reactor containment and upstream of the nonreturn valves and the swing disc isolation valves. Discharge from each of the safety valves is carried to atmosphere. The lowest safety valve set pressure is 1085 psia. The minimum total relieving capacity of all the valves is 10,068,845 lb per hour.

10.3.2.3 Main Steam Power-Operated Relief Valves

Three power operated relief valves (PORV) are provided which are capable of releasing heat to the atmosphere. These valves are automatically controlled by pressure or may be manually operated from the control board. In addition, in the event of a load rejection of greater than 70 percent, the steam dump controls can take over the operation of these valves. The PORV are together capable of releasing ten percent of the equivalent nominal rated steam flow (1,006,884 lb per hour of steam at 1020 psi pressure). One power operated relief valve, located on each main steam line upstream of the nonreturn valve and the swing disc isolation valve, is provided for each steam generator. Discharge from each of the three power operated relief valves is carried to the atmosphere.

The steam generator power operated relief valves provide the means for plant cooldown by steam discharge to the atmosphere if the condenser steam dump is not available. These relief valves are sized to pass a total steam flow of 10 percent of the plant maximum calculated flow using one valve on each steam generator line. The relief valves are of the modulating type, remote pressure controlled with remote adjusted relief pressure setting.

10.4 OTHER FEATURES OF THE STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN CONDENSER

10.4.1.1 Design Basis

The design parameters for the main condenser are shown in Table 10.4.1-1.

10.4.1.2 System Description

The condenser is the radial flow type with semicylindrical water boxes bolted at both ends. The hotwell is the deaerating type with storage sufficient for five minutes operation at maximum throttle flow with an equal free volume for surge protection. It has required manholes, a water gauge glass to indicate the condensate level, and two condensate outlets with coarse strainers. Expansion joints for all circulating water inlet and outlet connections are provided.

A condenser drain system is installed which provides a means to recirculate or empty the Unit 2 condenser hotwells. In the case of a primary to secondary steam generator leak, the contaminated condensate can be transferred to the waste condensate tanks.

10.4.1.3 Instrumentation Application

The more significant malfunctions or faults in the main condenser which cause alarms are:

a) Low vacuum in condenser
b) Low water level in condenser hotwell
c) High water level in condenser hotwell
d) High temperature in low pressure exhaust hood

10.4.8 AUXILIARY FEEDWATER SYSTEM

10.4.8.1 Design Basis

The design parameters for the auxiliary feedwater system components are shown on Table 10.4.8-1. The auxiliary feedwater system is designed and constructed in accordance with the Seismic Class I requirements presented in Section 3.2.

10.4.8.2 System Description

The flow diagram for the auxiliary feedwater system is included with the condensate and feedwater flow diagram Figures 10.1.0-4, 10.1.0-5, and 10.1.0-6.

Two auxiliary motor driven feedwater pumps using power from the plant diesel generators supply feedwater for decay heat removal during a complete loss of offsite power. A steam driven auxiliary feedwater pump also can supply feedwater to the steam generators during this period. Steam produced from decay heat removal drives the turbine.

The auxiliary feedwater pumps and turbine are supplied with bearing cooling water from the service water system.

The capacity of the steam driven auxiliary feedwater (AFW) pump is based on preventing the water level in the steam generators from receding below the lowest level within the indicated level range in the event of a loss of offsite power. This will prevent the tube sheet from being uncovered. A signal indicating a low steam generator water level or a direct signal of loss of power will automatically start the steam driven AFW pump by opening steam admission valves and auxiliary feedwater discharge valves to individual steam generators. The initiating signals for starting the motor driven AFW pumps are loss of voltage to both main feedwater pumps, low water level in any steam generator, or initiation of a safety injection signal. No operator action is required although at some time it may be desirable to manually trip the flow to the individual steam generators. Provision is made in the control room for control of the individual steam generator flow.

The steam supply to the steam driven AFW pump is taken off upstream from the main steam isolation valves, thereby assuring a source of steam to the pump. Main steam nonreturn valves (power operated stop-checks) are provided in each steam generator steam line. In the unlikely event of a steam line break the action of the nonreturn valve on the broken line prevents steam from the other two steam generators from discharging through the break. Feedwater to the unit with the ruptured line is isolated and the unit allowed to boil dry. The auxiliary feedwater pumps operation is automatic.

Should a steam line break occur in the header between the main steam isolation valves and the turbine, all main steam isolation valves are closed automatically. With the coincident loss of auxiliary power, emergency cooldown procedures are followed. If one main steam isolation valve fails to close, the feedwater line to the affected unit is isolated and the unit allowed to boil dry. Plant cooldown is then effected using the remaining two steam generators.

10.4.8.3 Safety Evaluation

In the unlikely event of complete loss of offsite electrical power to the station, decay heat removal would continue to be assured by the availability of one steam driven, and two motor driven auxiliary feedwater pumps, and steam discharge to atmosphere via the main steam safety valves and power operated relief valves. In this case, feedwater is available from the condensate storage tank by gravity feed to the auxiliary feedwater pumps. The 132,000 gallons of water in the condensate storage tank are adequate for decay heat removal for a period of at least twelve hours. Alternate sources of water are available from the lake via either leg of the plant service water system for an indefinite time period, and from the deep wells if offsite power is available.

If an auxiliary feedwater pump failed to start following a loss of main feedwater, sufficient redundance of feedwater pumps is available to provide the required feedwater. The steam driven AFW pump has twice the capacity of a motor driven emergency feedwater pump. One motor driven pump has sufficient capacity to prevent relief of fluid through the primary side relief valves.

The maximum starting time requirement for the motor driven auxiliary feedwater pumps is one minute. This allows sufficient time for the diesels to be started and auxiliary systems of higher priority to be loaded on the diesels. The motor driven auxiliary feedwater pumps are sized assuming the pumps are started within one minute including the allowance for starting the diesels and the loading sequence.

The maximum starting time requirement for the steam driven auxiliary feedwater pump is at most equal to that of the motor driven auxiliary feedwater pumps. This allows for starting delays and bringing the steam driven AFW pump to full load.

The adequacy of the auxiliary feedwater pump capacities is demonstrated in the Loss of Normal Feedwater Accident of Section 15.2.7.

The analysis of the effects of loss of full or partial load on the reactor coolant system is discussed in Section 15.2.2.

In the event of a failure of Lake Robinson Dam, shutdown would be accomplished in an orderly manner using the condensate storage tank. When the condensate storage tank reaches a low level limit, auxiliary feedwater pump suction would be changed to the deepwell pump discharge. This source would provide the required feedwater indefinitely or until such time some other source of feedwater can be established. It is assumed that emergency power is not required for this accident.

After boration is complete, the plant can be slowly cooled down by allowing the decay heat to be removed through the steam generator utilizing the steam driven auxiliary feedwater pump. This pump can be operated without external seal cooling water. With the minimum volume of water available in the condensate storage tank (35,000 gallons), the plant could be maintained at hot shutdown for at least two hours before makeup from the deep well pumps would be required.

Due to the location of the auxiliary feedwater steam driven and motor driven pumps inside buildings, reasonable assurance is provided that no loss of function of the system will occur because of tornado damage. In the event the steam lines supplying steam to the turbine driven pump are damaged, the motor driven pump, powered by the emergency diesel generators located in Auxiliary Building, can be used.

10.4.8.4 Tests and Inspections

The auxiliary feedwater pumps can be periodically operated to verify their operability during plant shutdown.

Proper functioning of the steam admission valve and subsequent starting of the steam driven AFW pump demonstrate the integrity of the system. Verification of correct operation can be made both from instrumentation within the main control room and direct visual observation of the pump.

10.4.8.5 Instrumentation Requirements

The controls used to automatically start the auxiliary feedwater pumps are designed to meet the single failure criterion. The following pump starting logic is used:

a) The two motor driven auxiliary feedwater pumps are started automatically on:

1) 2/3 low low level in any steam generator

2) Opening of both feedwater pump circuit breakers (one contact per pump breaker is used)

3) Any safety injection signal

4) Loss of all AC power (i.e., the blackout sequence)

5) Manually.

b) The steam driven auxiliary feedwater pump is started automatically on:

1) 2/3 low low level in any two steam generators

2) Loss of voltage on both 4.16 kV buses 1 and 4. Two sensors are provided for each bus with 1/2 logic to indicate a loss of voltage on any one bus

3) Manually.

The relay logic for starting the auxiliary feedwater pumps is separated into train A and train B logic as is done for the relay logic used to actuate engineered safety features equipment. Logic train A will start one motor driven pump and logic train B will start the second motor driven pump. Either logic train will open appropriate steam system valves to start the turbine driven pump. The circuits used to start the auxiliary feedwater pumps will also open the appropriate valves to ensure delivery of flow to the steam generators. The failure analysis for the auxiliary feedwater system is shown in Table 10.4.8-2.

CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

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CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

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11.2.2 SYSTEM DESCRIPTION

11.2.2.1 Liquid Radioactive Waste Processing

During normal plant operation the WDS processes liquids from the following sources:

a) Equipment drains and leakoffs

b) Radioactive chemical laboratory drains

c) Radioactive shower drains

d) Decontamination area drains

e) Demineralizer regeneration

As shown in the flow diagrams, Figures 11.2.2-1 through 11.2.2-4, the system also collects and transfers liquids from the following sources directly to the CVCS for processing:

a) Reactor coolant loop drains

b) Pressurizer relief tank

c) Reactor coolant pump secondary seals

d) Excess letdown during startup

e) Accumulators

f) Valve and reactor vessel flange leakoffs

These liquids flow to the reactor coolant drain tank and are discharged directly to the CVCS holdup tanks by the reactor coolant drain pumps which are operated automatically by a level controller in the tank. These pumps also return water from the refueling canal and cavity to the refueling water storage tank.

Where possible, waste liquids drain to the waste holdup tank by gravity flow. Other waste liquids drain to the sump tank and are discharged to the waste holdup tank by pumps operated automatically by a level controller in the tank.

Since the radioactivity level of waste liquid from the hot shower area will usually be low enough to permit discharge from the site without processing, two tanks are provided to permit one tank to be filled and isolated for sampling and analysis while the second is in service. If preliminary analysis indicates that the liquid is suitable for discharge, it is pumped to one of the waste condensate tanks where its radioactivity can be determined for record before it is discharged through a radiation monitor to the condenser circulating water. Otherwise, the liquid is pumped to the waste holdup tank or through the polishing demineralizers for processing. Similar capability is provided to bypass the low level waste from the waste holdup tank around the waste evaporator. Although the waste holdup tank cannot be isolated while the

preliminary analysis is made, the final analysis for record is done in the waste condensate tanks which can be isolated to ensure no added contamination during sampling, analysis, and release.

Liquids requiring cleanup before release are processed by recycling the contents of the waste condensate tanks through the polishing demineralizers or in batches by the waste evaporator. The concentrated bottoms are discharged to the drumming room where they are processed for removal to a burial facility. The condensate is routed to one of the waste condensate tanks. When one tank is filled, it is isolated and sampled for analysis while the second tank is in service. If analysis confirms the activity level is suitable for discharge, the condensate is pumped through a flow meter and a radiation monitor to the condenser circulating water discharge. Otherwise it is returned to the waste holdup tank for reprocessing or recirculated through the polishing demineralizers. Although the radiochemical analysis forms the basis for recording activity releases, the radiation monitor provides surveillance over the operation by closing the discharge valve if the liquid activity level exceeds a preset valve.

11.2.2.2 Components

Codes applying to components of the WDS are shown in Table 3.2.2-5. Components summary data are shown in Table 11.2.1-1.

Laundry and Hot Shower Tanks - Two stainless steel tanks collect liquid wastes originating from the hot shower. When a tank has been filled, its contents are analyzed for gross β-γ activity. The tank contents may be drained to the waste condensate tanks or to the waste holdup tanks for waste evaporator processing.

Chemical Drain Tank - The stainless steel chemical drain tank is provided to collect drainage from the hot area of the chemistry laboratory. After analysis, the tank contents are pumped to the waste holdup tanks or to the waste condensate tanks.

Reactor Coolant Drain Tank - The reactor coolant drain tank is all-welded austenitic stainless steel. This tank serves as a drain collecting point for the RCS and other equipment located inside the reactor containment.

Waste Holdup Tank - The waste holdup tank retains radioactive liquids from the CVCS, sump tank, chemical drain tank, reactor coolant drain tank, and laundry and hot shower tanks. The tank is stainless steel of welded construction.

Sump Tanks and Pumps - The sump tanks serve as collecting points for waste discharged to the basement level drain header. They are located at the lowest point in the Auxiliary Building. All floor drains entering these tanks contain loop seals to prevent gas from leaving the pressure vent system. Two horizontal centrifugal sump pumps drain these tanks. All wetted parts of the pumps are stainless steel. The tanks are all welded austenitic stainless steel.

Spent Resin Storage Tank - The spent resin storage tank retains spent resin normally discharged from the mixed bed, spent fuel pit, base removal, and cation and polishing demineralizers. The deborating demineralizers and evaporator condensate demineralizers are regenerated and therefore will be discharged less frequently. Normally, the tank is filled over a long period of time, the contents are allowed to decay, and then emptied prior to receiving any additional resin. However, the contents can be removed at any time, if sufficient shielding is provided for the spent resin shipping vessel. A layer of water is maintained over the resin surface to prevent resin degradation due to heat generation from decaying fission products. Resin is removed from the tank by first backflushing with nitrogen to loosen the resin bed and then flushing the resin to the drumming room with nitrogen entering the top of the tank. The tank is all welded austenitic stainless steel.

Waste Evaporator Package - The evaporator concentrates dissolved and suspended solids in the liquid waste. It consists of a batch tank (feed), concentrator, distillate tank, hot water converter, batch tank pump, hot water circulating pump, distillate pump, and control panel.

The length of an evaporator operating cycle is determined either by solids content or activity concentration of the solution. The entire evaporator is austenitic stainless steel of welded construction except for the heat transfer surfaces which are admiralty metal.

"A" and "B" waste condensate tanks are located in the Auxiliary Building in the hallway just outside the pipe alley entrance. These two tanks receive distillate from "A" waste evaporator and provide a temporary storage for this distillate water until it can be released to the environment.

"C", "D", and "E" waste condensate tanks receive distillate from "B" waste evaporator and they are located just north of the Auxiliary Building. When a waste condensate tank gets to approximately 90 percent level, the distillate from the evaporator is aligned to go to an alternate tank. The "full" tank can then be recirculated and sampled for release.

Pumps - Pumps used throughout the system for draining tanks and transferring liquids are shown in Figures 11.1-1 and 11.1-2. The pumps are either canned motor or mechanically sealed types to minimize leakage.

The wetted surfaces of all pumps are stainless steel or other materials of equivalent corrosion resistance.

Piping - The piping which carries liquid wastes is stainless steel. All gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance. No threaded fittings are used in waste piping.

Valves - All valves exposed to gases are carbon steel. All other valves are stainless steel. All valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage.

Stop valves are provided to isolate equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive wastes if the tanks might be overpressurized by improper operation or component malfunction.

Concentrates Holding Tank - The concentrates holding tank is sized to hold the production of concentrates from one batch of evaporator operation. The tank is supplied with an electrical heater which prevents boric acid precipitation and is constructed of austenitic stainless steel.

11.2.3 RADIOACTIVE RELEASES

11.2.3.1 Liquid Effluent Source Terms

Liquid effluents from HBR 2 will occur both continuously and on a batch basis. The following sections discuss the methodology which is utilized to show compliance with 10CFR20.

Continuous Releases - Steam generator (SG) blowdown is continuously released from HBR during normal operation. A daily grab sample is taken of SG blowdown. This sample is composited and analyzed weekly for I-131 and various other fission, activation, and corrosion products. Compliance with 10CFR20 during actual release is established through the SG blowdown effluent monitor alarm setpoint. However, if a continuous release should occur in which the effluent monitor alarm setpoint is exceeded, then actual compliance with 10CFR20 may be determined utilizing the actual radionuclide mix.

Batch Releases - Batch releases occur during normal operation. When this occurs at HBR, a continuous release is usually also occurring. However, during cold shutdown conditions only batch releases may occur at HBR. The radioactivity content of each batch release will be determined prior to release and will show compliance with 10CFR20.

For HBR, the liquid radwaste system discharges to the circulating water system. Therefore, the dilution flow rate (DFR) is a function of the number of circulating water pumps operating. HBR 2 has 3 circulating water pumps. Pump curves show that with 3 pumps operating, the circulating water flow rate is 400,000 gpm, with 2 pumps operating the flow rate is 250,000 gpm, and with 1 pump, the flow rate is 160,700 gpm. At least one circulating water pump must be operating during any liquid waste discharge.

Batch releases from the HBR liquid radwaste system may occur from the waste condensate tanks, the monitor tanks, and the steam generator (SG). The maximum release rate is 300 gpm from the steam generator and 60 gpm from the monitor and waste condensate tanks.

As discussed in Section 11.5, the Steam Generator Liquid Sample Monitor (RMS-19) is limited to 50 percent of 10CFR20 requirements and the WDS Liquid Effluent Monitor (RMS-18) setpoint is limited to 80 percent of the 10CFR20 limits. These setpoints ensure that 10CFR20 limits are met. However, because they are based upon a given mix, the possibility exists that the alarm trip setpoints may be exceeded and 10CFR20 limits may not actually be exceeded.

Liquid wastes are generated primarily by plant maintenance and service operations. Considerable operational margin exists between the estimated load on the waste disposal system and the design capability.

11.2.3.2 Doses from Liquid Effluents

The dose contribution from the release of liquid effluents is calculated once per 31 days and a cumulative summation of these total body and organ doses is maintained for each calendar quarter.

The sum of the cumulative doses from all batch and continuous releases for a quarter are compared to one half the design objective doses for total body and any organ. The sum of the cumulative doses from all batch and continuous releases for a calendar year are compared to the design objective doses.

For the calendar quarter,

$$D_\tau \leq 1.5 \text{ mrem total body} \quad (1)$$

$$D_\tau \leq 5 \text{ mrem any organ} \quad (2)$$

For the calendar year,

$$D_\tau \leq 3 \text{ mrem total body} \quad (3)$$

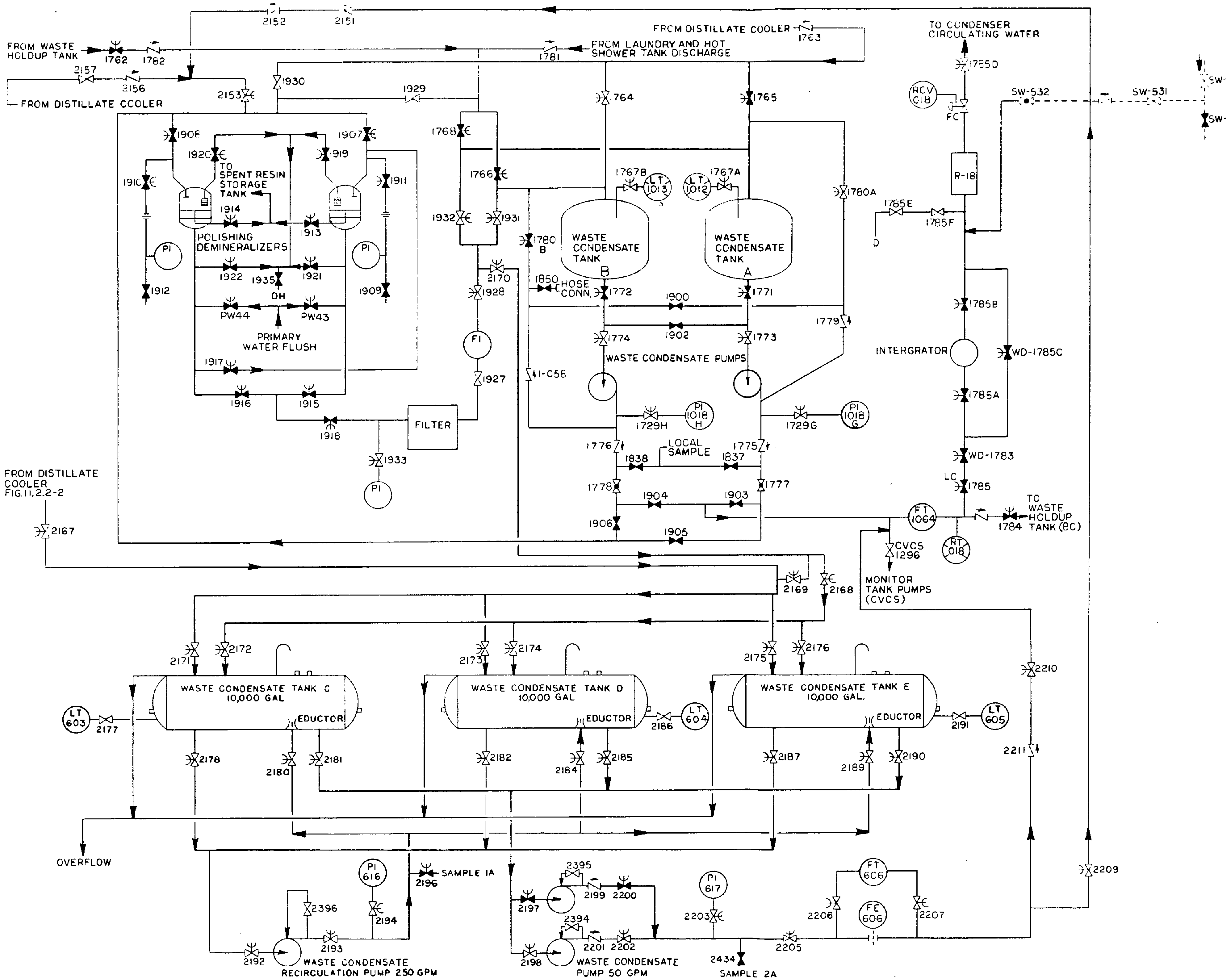
$$D_\tau \leq 10 \text{ mrem any organ} \quad (4)$$

where: D_τ = cumulative total dose to any organ τ or to the total body from continuous and batch releases, mrem

$= D_{\tau b} + D_{\tau c}$

The quarterly limits given above represent one-half the annual design objective of Section II.A of Appendix I of 10CFR50. If any of the limits are exceeded, a special report pursuant to Section IV.A of Appendix I of 10CFR50 must be filed with the NRC.

A summation of all liquid releases, along with offsite dose calculation results, is documented in the Effluent and Waste Disposal Semi-Annual Report submitted to the NRC semi-annually.



Ref. Dwg. CP-406 5379-920 Sheet 2 Rev. 13

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UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM
LIQUID WASTE DISPOSAL SYSTEM
SHEET 3

FIGURE 11.2.2 - 3

11.3.2 SYSTEM DESCRIPTION

11.3.2.1 Gaseous Radioactive Waste Processing

During plant operations, gaseous wastes will originate from:

a) Degassing reactor coolant discharged to the Chemical and Volume Control System (CVCS)

b) Displacement of cover gases as liquids accumulate in various tanks

c) Miscellaneous equipment vents and relief valves

d) Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases

The flow diagrams for the system are shown in Figures 11.3.2-1 and 11.3.2-2.

Radioactive gases are collected at a slight positive pressure in a vent header. From there, they are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with holdup capacity and discharge controls for gaseous wastes such that plant operations will not be limited by environmental conditions.

During normal operation the Waste Disposal System supplies nitrogen and hydrogen from standard cylinders to primary plant components. This operation is identical for both the nitrogen supply and the hydrogen supply. For each, two headers are provided, one for operation and one for backup. The pressure regulator in the operating header is set for 100 psig discharge and that in the backup header for 90 psig discharge. When the operating header is exhausted, its discharge pressure will fall below 100 psig and an alarm will alert the operator. The second bank will come into service automatically at 90 psig to ensure a continuous supply of gas. After the exhausted header has been recharged, the operator manually sets the operating pressure back to 100 psig and the backup pressure at 90 psig.

Most of the gas received by the Waste Disposal System during normal operation is cover gas displaced from the CVCS holdup tanks as they fill with liquid. The pressure of cover gases are maintained within a narrow range. As the tanks are filled displacing cover gas, the pressure rises. When the upper limit of the range is approached, the waste gas compressors pump the displaced gas to the gas decay tanks. As the tanks are emptied, the cover gas pressure approaches the lower limit of the range and additional gas is supplied. Since the gas displaced during filling must be replaced when the tanks are emptied, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. To prevent hydrogen concentration from exceeding the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or aerated liquids and the vent header itself is

designed to operate at a slight positive pressure (0.5 psig minimum to 2.0 psig maximum) to prevent in-leakage. On the other hand, out-leakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self contained pressure regulators and soft-seated packless valves throughout the radioactive portions of the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first unit. From the compressors, gas flows to one of the gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select one tank for backup if the tank in operation becomes fully pressurized. When the tank in service becomes pressurized to 110 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank and sounds an alarm to alert the operator of this event so that he may select a new backup tank. Pressure indicators are supplied to aid the operator in selecting the backup tank.

Gas held in the decay tanks can either be returned to the CVCS holdup tanks, or discharged to the atmosphere if it has decayed sufficiently for release. Generally, the last tank to receive gas will be the first tank emptied back to the holdup tanks in order to permit the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator freedom to fill, reuse or discharge gas to the environment simultaneously without restricting operation of the other tanks. During degassing of the reactor coolant prior to a refueling shutdown, it may be desirable to pump the gas purged from the volume control tank into a particular tank and isolate that tank for decay rather than reuse the gas in it. This is done by aligning the control to open the inlet valve to the desired tank and closing the outlet valve to the reuse header. However, one of the other tanks can be opened to the reuse header at this time if desired, while still another might be discharged to atmosphere.

Before a tank can be emptied to the environment, it must be sampled and analyzed to determine and record the activity to be released, and only then discharged to the plant vent at a controlled rate through a radiation monitor. Samples are taken manually by opening the isolation valve to the plant vent discharge line and permitting gas to flow to the gas analyzer where it can be collected in one of the sampling system gas sample vessels. After sampling, the isolation valve is closed until the tank contents are released. During release a trip valve in the discharge line is closed automatically by a high activity level indication in the plant vent.

When waste gases are being released to the environment, the release is automatically terminated if the radioactivity level exceeds a predetermined level (the radiological monitoring and control instrumentation is described in Section 11.5).

During operation, gas samples are drawn periodically from tanks discharging to the waste gas vent header as well as from the particular gas decay tank being filled at the time, and automatically analyzed to determine their hydrogen and oxygen content. The hydrogen analysis is for surveillance since the concentration range will vary considerably from tank to tank. There should be

no significant oxygen content in any of the tanks, and an alarm will warn the operator if any sample shows 2 percent by volume of oxygen. This allows time to take the required action before the combustible limit is reached. Another tank is placed in service while the operator locates and eliminates the source of oxygen.

11.3.2.2 Components

Component data are given in Table 11.3.2-1.

Nitrogen Manifold

A dual manifold supplies nitrogen to purge the vapor space of various components to reduce the hydrogen concentration or to replace fluid that has been removed. A pressure controller, which automatically switches from one manifold to the other, assures a continuous supply of gas.

Hydrogen Manifold

A dual manifold supplies hydrogen to the volume control tank to maintain the hydrogen partial pressure as hydrogen dissolves in the reactor coolant. A pressure controller, which automatically switches from one manifold to the other, assures a continuous supply of gas.

Gas Analyzer

An automatic gas analyzer is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of the Waste Disposal System, CVCS tanks, boric acid evaporators and gas stripper. Upon indication of a high oxygen level, provisions are made to purge the equipment to the gaseous waste system with an inert gas.

Piping

All gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance. No threaded fittings are used in waste piping.

Valves

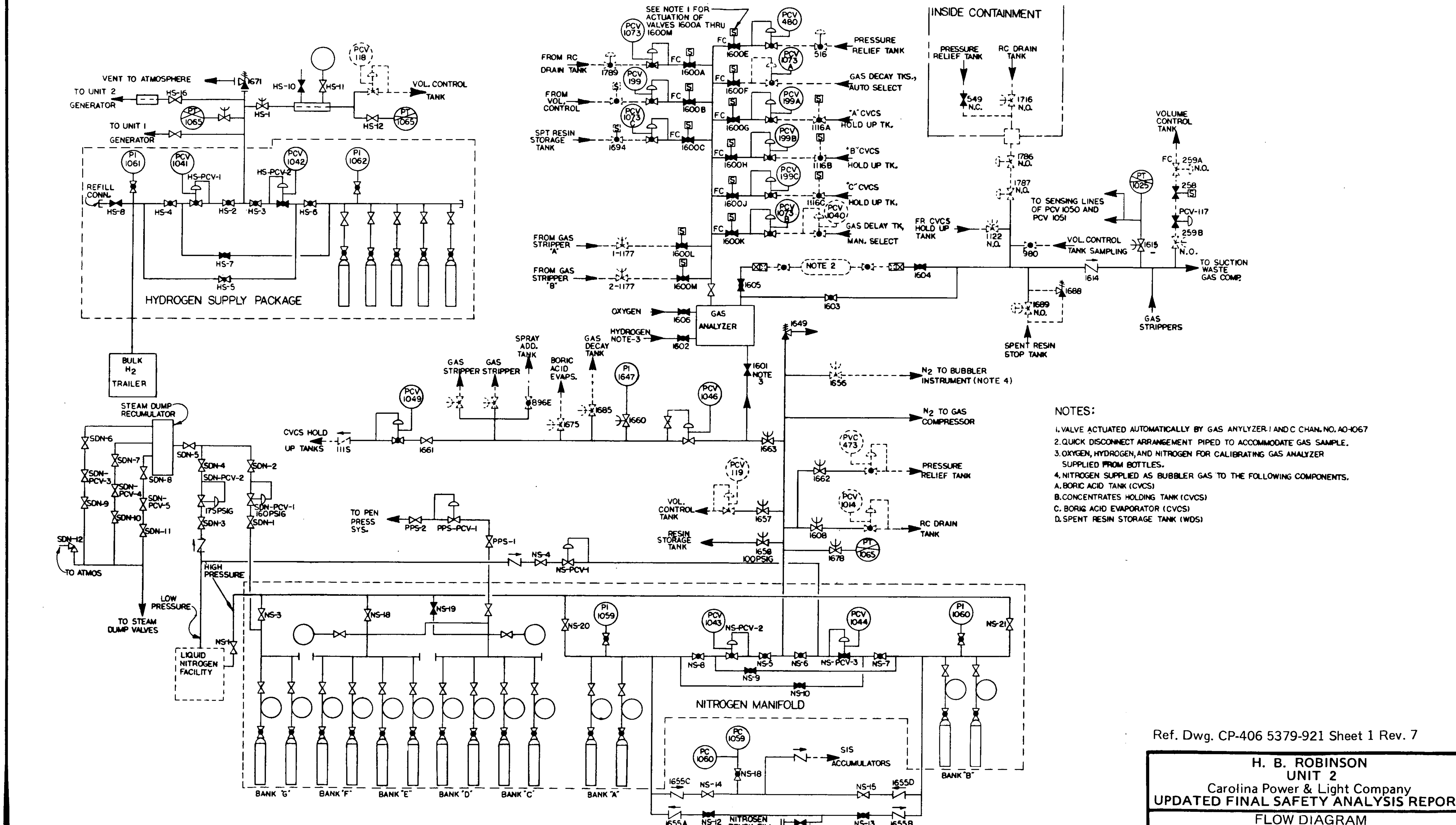
All valves exposed to gases are carbon steel. All valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage.

Gas Compressors

Two compressors are provided for removal of gases to the gas decay tanks from all equipment that contains or can contain radioactive gases. These compressors are of the water sealed centrifugal displacement type. The operation of the compressors is automatically controlled by the gas manifold pressure. While one unit is in operation, the other serves as a standby for unusually high flows or failure of the first unit.

Gas Decay Tanks

Four welded carbon steel tanks are provided to contain compressed waste gases (hydrogen, nitrogen, oxygen and fission gases). After a period for radioactive decay, these gases may be released at a controlled rate to the atmosphere through the plant vent. All discharges to the atmosphere will be monitored.



Ref. Dwg. CP-406 5379-921 Sheet 1 Rev. 7

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FLOW DIAGRAM
GASEOUS WASTE DISPOSAL SYSTEM
SHEET 1

FIGURE 11.3.2 - 1

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 DESIGN BASIS

The Process and Effluent Radiological Monitoring Systems are designed to:

a) Warn of any inadvertant release of radioactive effluents to the environment

b) Give early warning of a plant malfunction which might lead to a health hazard or plant damage.

Instruments are located at selected points in and around the plant to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is activated in the Control Room. The automatic Radiation Monitoring System (RMS) operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning are thereby provided for the continued safe operation of the plant and compliance with 10CFR20 and NUREG 0578 Section 2.18b.

The only components of these systems which are located in the containment are the detectors for certain area monitoring channels. They would not be expected to operate following a major loss-of-coolant accident (LOCA) and are not designed for this purpose. Components of all other area and process monitoring channels are designed for post-accident operation.

The components of the RMS are designed according to the following environmental conditions:

a) Temperature - an ambient temperature range of 40 to 120^{o}F

b) Humidity - 0 to 100 percent relative humidity

c) Pressure - Components in the Auxiliary Building and Control Room are designed for normal atmospheric pressure

d) Radiation - Process and area radiation monitors are of a nonsaturating design so that they "peg" full scale if exposed to radiation levels up to 100 times full scale indication. Process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Components

This system consists of ten channels which monitor radiation levels in various plant operating systems. The output from each channel detector is transmitted to the RMS cabinets located in the Control Room area where the radiation level is indicated by a meter and recorded by a multipoint recorder. High radiation level alarms are annunciated in the Control Room and indicated on the RMS cabinets.

Each channel contains a complete integrated modular assembly, which includes the following:

a) Level Amplifier - amplifies the energy of the radiation pulse to provide a discriminated output to the log level amplifier.

b) Log Level Amplifier - accepts the shaped pulse of the level amplifier output, performs a log integration (converts total pulse rate to a logarithmic analog signal), and amplifies the resulting output for suitable indication and recording.

c) Power Supplies - power supplies are contained in each drawer for furnishing the positive and negative voltages for the transistor circuits, relays and alarm lights, and for providing the high voltage for the detector. In the event of a power outage, a backup power supply for these channels is available (see Section 8.3).

d) Test-Calibration Circuitry - these circuits provide a precalibrated analog signal to perform channel test, and a solenoid operated radiation check source to verify the channel's operations. An annunciator light on the reactor and turbine-generator (RTG) board indicates when the channel is in the test calibrate mode.

e) Radiation Level Meter - this meter, mounted on the drawer, has a scale calibrated logarithmically in counts per minute from 10^1 to 10^4, and 10^1 to 10^6. The level signal is also recorded by the recorder.

f) Indicating lights - these lights indicate high-radiation alarm levels and circuit failure. An annunciator on the RTG board is actuated on high radiation.

g) Bi-stable Circuits - two bi-stable circuits are provided, one to alarm on high radiation (actuation point may be set at any level over the range of the instruments), and one to alarm on loss of signal (circuit failure).

h) A remotely operated long half-life radiation check source is furnished in each channel. The energy emission ranges are similar to the radiation energy spectrá being monitored. The source strength is sufficient to cause a positive indication of the detector unit.

11.5.2.2 Process and Effluent Radiation Monitoring System

The process and effluent RMS consists of the radiation monitoring channels described below. Table 11.5.2-1 lists the detecting medium conditions for each channel, and Table 11.5.2-2 gives their sensitivites.

11.5.2.2.1 Containment or Plant Vent Air Particulate Monitor (RMS-11)

The monitor is provided to measure air particulate gamma radioactivity in the containment and to ensure that the release rate through the containment vent during purging is maintained below specified limits. It also provides a backup to the plant vent gas monitor. High radiation level for the channel initiates closure of the containment purge supply and exhaust duct valves and pressure relief line valves. This monitor has a measuring range of 10^{-9} to 10^{-6} μCi/cc.

This channel takes a continuous air sample from either the containment atmosphere, or the plant vent. The sample is drawn from the containment or the plant vent ductwork through a closed, sealed system monitored by a scintillation counter - filter paper detector assembly. The filter paper collects all particulate matter greater than 1 micron in size, on its constantly moving surface, and is viewed by a photomultiplier-scintillation crystal combination. The sample is returned to the containment or plant vent, depending on which sample is being monitored, after it passes through the series connected gas monitor (RMS-12).

The detector assembly is in a completely enclosed housing. The detector is a hermetically sealed photomultiplier tube - beta phosphor scintillation combination. A preamplifier transmits the pulse signal to the RMS cabinets in the Control Room. The filter paper has a 25-day minimum supply at normal speed. Lead shielding is provided to reduce the background level to where it does not interfere with the detector's sensitivity. The filter paper mechanism, an electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

11.5.2.2.2 Containment and Plant Vent Radioactive Gas Monitor (RMS-12)

One monitor is provided to measure radioactivity from noble gases in the containment, to ensure that the radiation release rate during purging is maintained below specified limits and to serve as a backup to the plant vent gas monitor. High gas radiation level initiates closure of the containment purge supply and exhaust duct valves and pressure line relief valves.

This monitor has a measuring range of 10^{-6} to 10^{-3} μCi/cc.

This channel takes the continuous air sample from the containment atmosphere or the plant vent after it passes through the air particulate monitor, and draws the sample through a closed, sealed system to the gas monitor assembly. The sample is constantly mixed in the fixed, shielded volume, where it is viewed by the Geiger-Mueller (GM) tube. The sample is then returned to the containment or the plant vent depending on which sample is being monitored.

11.5.2.2.6 Component Cooling Liquid Monitor (RMS-17)

This channel continuously monitors the component cooling loop of the Auxiliary Coolant System for activity indicative of a leak of reactor coolant from either the Reactor Coolant System, the recirculation loop or the residual heat removal loop of the Auxiliary Coolant System. A scintillation counter is located in an in-line well at the component cooling pump suction header. The detector assembly output is amplified by a preamplifier and transmitted to the RMS cabinets in the Control Room. The activity is indicated on a meter and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator in addition to the RMS cabinets. A high-radiation level alarm signal initiates closure of the valve located in the component cooling surge tank vent line to prevent gaseous radiation release. The measuring range of this monitor is 10^{-5} to 10^{-2} μc/cc.

11.5.2.2.7 Waste Disposal System Liquid Effluent Monitor (RMS-18)

This channel continuously monitors all Waste Disposal System liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. A scintillation counter and holdup tank assembly monitors these effluent discharges. Remote indication and annuciation are provided on the Waste Disposal System control board. The measuring range of this monitor is 10^{-5} to 10^{-2} μCi/cc.

11.5.2.2.8 Steam Generator Liquid Sample Monitor (RMS-19)

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air removal gas monitor. Samples from the bottom of each of the steam generators are mixed in a common header and the common sample is continuously monitored by a scintillation counter and holdup tank assembly. Upon indication of a high-radiation level, each steam generator is individually sampled in order to determine the source. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 min).

A remote indicator panel, mounted at the detector location, indicates the radiation level and high-radiation alarm.

At levels above 1.3×10^{-5} μCi/ml high-radiation level signal will close the isolation valves in the blowdown lines and sample lines. This setpoint is about 10 percent of the Technical Specification limits for the release rate of unidentified radionuclides to the lake and to the atmosphere due to particulates and iodine with greater than 8-day half-life. The measuring range of this monitor is 10^{-5} to 10^{-2} μCi/cc.

In channels RMS-16, RMS-18, and RMS-19, a photomultiplier tube-scintillation crystal (NaI) combination, mounted in a hermetically sealed unit, is used for liquid effluent radiation actuation. Lead shielding is provided to reduce the background level so it does not interfere with detector's sensitivity. The in-line fixed volume container is an integral part of the detector unit.

In addition, grab samples from the monitor tanks are analyzed to determine the amount of tritium in the effluent discharged to the Waste Disposal System. During normal plant operation, grab samples from the containment and Auxiliary Building areas are analyzed for tritium as required.

11.5.2.2.9 Fuel Handling Building Basement Exhaust Monitor (RMS-20)

This channel consists of three monitors. The first monitor, located near the point of atmospheric discharge, continuously monitors the radio noble gases released through this exhaust vent. This monitor's sensitivity ranges from 3×10^{-6} μCi/cc to 1×10^{-1} μCi/cc.

The second monitor, located in the same monitor housing as the first, provides intermittent particulate/iodine sampling.

The third monitor, located adjacent to the basement exhaust duct, is a high range noble gas monitor able to releases of the magnitude of 10,000 Ci/sec (2.1×10^{-3} μCi/cc).

Each of these monitors is set to alarm if a gaseous release rate equal to 10 percent of that permitted by the Technical Specifications is reached. If the alarm point is reached, the ventilation system automatically shuts down, thus ending any release to the environment.

11.5.2.2.10 Fuel Handling Building Upper Level Exhaust Vent (RMS-21)

This channel consists of one radio noble gas monitor which continuously monitors releases through this exhaust vent. If the predetermined alarm point is exceeded, the ventilation system automatically shuts down, thus ending any release to the environment.

11.5.2.2.11 Plant Vent Iodine, Particulate, and Noble Gas Monitors

These monitors continuously sample the plant vent for iodine, particulate, and noble gas activity. These monitors are combined into a single three channel system. This monitoring system is designed to measure beta-plus-gamma activity detectable from airborne dust, radioactivity from iodine vapor present in the air after filtration, and beta-gamma activity residual in the gas after particulate and iodine filtering. The particulate monitor detects and measures the radioactivity collectable from the air by specially selected filter paper. The particulate monitor utilizes an SC-2B beta scintillation detector. The iodine monitor makes use of an activated charcoal cartridge for concentrating the iodine, with an SC2-1S gamma scintillation detector measuring the buildup of iodine activity. After particulate and iodine filters, the residual gas is passed through a heavily shielded one liter chamber where the noble gas monitor, a halogen quenched, GM-912, Geiger-Mueller tube measures both beta and gamma activity. These three monitors operate into a three channel counting ratemeter which in turn controls a three channel graphic recorder. The particulate and iodine filters from these monitors are removed and counted in the counting room on a regular schedule and serve as the permanent record for offsite releases of particulate and iodine activity. Recorded readings from the noble gas monitor are used to calculate the permanent record for offsite releases of noble gas activity.

TABLE 11.5.2-1

DETECTING MEDIUM CONDITIONS

CHANNEL	MEDIUM	TEMPERATURE RANGE (^{o}C)
RMS-11	Air	10-50
RMS-12	Air	10-50
RMS-14	Air	4-50
RMS-15	Air	10-50
RMS-16	Water	15-71
RMS-17	Water	4-71
RMS-18	Water	15-71
RMS-19	Water	15-71
RMS-20	Air	10-50
RMS-21	Air	10-50

TABLE 11.5.2-2

RADIATION MONITORING SYSTEM CHANNEL SENSITIVITIES

CHANNEL	SENSITIVITY RANGE (μCi/cc)	DETECTED ISOTOPES
RMS-11	1.0×10^{-9} to 1.0×10^{-6}	I^{131}, I^{133}, Cs^{134}, Cs^{137}
RMS-12	1.0×10^{-6} to 1.0×10^{-3}	Kr^{85}, Ar^{41}, Xe^{135}, Xe^{133}
RMS-14	5.0×10^{-7} to 5.0×10^{-4}	Kr^{85}, Ar^{41}, Xe^{135}, Xe^{133}
RMS-15	1.0×10^{-6} to 1.0×10^{-3}	Kr^{85}, Ar^{41}, Xe^{135}, Xe^{133}
RMS-16	1.0×10^{-5} to 1.0×10^{-2}	Co^{60}, Mixed Fission Products
RMS-17	1.0×10^{-5} to 1.0×10^{-2}	Co^{60}, Mixed Fission Products
RMS-18	1.0×10^{-5} to 1.0×10^{-2}	Co^{60}, Mixed Fission Products
RMS-19	1.0×10^{-5} to 1.10×10^{-2}	Co^{60}, Mixed Fission Products
RMS-20	3.0×10^{-6} to 2.1×10^{-3}	Kr^{85}, Ar^{41}, Xe^{135}, Xe^{133}, I^{131}, I^{133}, CS^{134}, CS^{137}
RMS-21	1.0×10^{-6} to 1.0×10^{-3}	Kr^{85}, Ar^{41}, Xe^{135}, Xe^{133}

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RADIATION PROTECTION

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RADIATION PROTECTION

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CHAPTER 12

RADIATION PROTECTION

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12.3.3 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The Radiation Monitoring System (RMS) consists of the following:

a) Area Radiation Monitoring System

b) Process and Effluent Radiological Monitoring and Sampling System

The Area Radiation Monitoring System is described below, while the Process and Effluent Monitoring and Sampling System (which includes Airborne Radiation Monitoring) is described in Section 11.5.

12.3.3.1 Area Radiation Monitoring System

12.3.3.1.1 Design Basis

The Radiation Monitoring System is designed to perform two basic functions:

a) Warn of any radiation health hazard which might develop

b) Give early warning of a plant malfunction which might lead to a health hazard or plant damage

Instruments are located at selected points in and around the plant to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the Control Room. The automatic Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning are thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed 10CFR20 limits.

The only components of these systems which are located in the containment are the detectors for certain area monitoring channels. They would not be expected to operate following a major loss-of-coolant accident (LOCA) and are not designed for this purpose. Components of all other area and process monitoring channels are designed for post-accident operation.

The components of the Radiation Monitoring System are designed according to the following environmental conditions:

a) Temperature - an ambient temperature range of 40° to 120°F

b) Humidity - 0 to 100 percent relative humidity

c) Pressure - Components in the Auxiliary Building and Control Room are designed for normal atmospheric pressure. Area Monitoring System components inside the containment are designed to withstand test pressure.

d) Radiation - Process and area radiation monitors are of a nonsaturating design so that they "peg" full scale if exposed to radiation levels up to 100 times full scale indication. Process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

12.3.3.1.2 General System Description

The Area Radiation Monitoring System consists of eight channels which monitor radiation levels in various areas of the plant. These areas are as follows:

CHANNEL	AREA MONITOR
R-1	Control Room
R-2	Containment
R-3	Radiochemistry Laboratory
R-4	Charging Pump Room
R-5	Spent Fuel Building
R-6	Sampling Room
R-7	In-Core Instrumentation Cubicle
R-8	Drumming Station

Each channel has a fixed position gamma sensitive GM tube detector. The detector output is amplified and the log count rate is determined by the integral amplifier at the detector. The radiation level is indicated locally at the detector, and at the Radiation Monitoring System cabinets where it is also recorded. High-radiation alarms are displayed on the RTG board, the radiation monitoring cabinets, and at the detector location.

The Control Room annunciator provides a single window which alarms for any channel detecting high-radiation. Verification of which channel has alarmed is done at the Radiation Monitoring System cabinets. A remotely operated, long half-life radiation check source is provided in each channel. The source strength is sufficient to produce approximately mid-range indication. The electronics module located in the Control Room cabinets amplifies the radiation level signal, computed by the log level preamplifier, for indication and recording. The module also provides controls for actuation of the channel check source.

A meter is mounted on the front of each computer indicator module and is calibrated logarithmically from 0.1 mR/hr to 10,000 mR/hr.

A remote meter, calibrated logarithmically from 0.1 mR/hr to 10,000 mR/hr is mounted at the detector assembly.

Radiation Monitoring System cabinet alarms consist of a red indicator light for high radiation and an amber light to annunciate detector or circuit failure. The remote meter and alarm assembly at the detector contains a red indicator light and a buzzer type alarm annunciator actuated on high radiation.

All of the Control Room system equipment is centralized in three cabinets. High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies, and entire channels. It is possible to completely remove the various chassis from the cabinet, after disconnecting the cables from the rear of these units.

A 24-point strip chart recorder is provided in the Radiation Monitoring System cabinets in the Control Room. Each monitoring channel is sequentially recorded. The print rate is 20 seconds per point. Chart speed is adjustable.

The sensitivity of the radiation monitors is 0.1 to 10^4 mR/hr.

Table 12.3.3-1 indicates the detector medium and temperature conditions during normal operation.

The relation of the radiation monitoring channels to the systems with which they are associated is given in the sections describing those systems. Routine test and recalibrations will ensure that the channels operate properly.

TABLE 12.3.3-1

DETECTING MEDIUM CONDITIONS

CHANNEL	MEDIUM	TEMPERATURE RANGE (^{o}C)
R-1	Air	10-50
R-2	Air	10-50
R-3	Air	10-50
R-4	Air	10-50
R-5	Air	10-50
R-6	Air	10-50
R-7	Air	10-50
R-8	Air	10-50

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

12.5.2.1 Personnel Protective Equipment

All personnel entering the Radiation Control Area are required to wear protective clothing. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. Some of the protective apparel available are shoe covers, head covers, gloves, coveralls, etc. Additional items of specialized apparel such as plastic or rubber suits, and respirators are available for operations involving wet surfaces or airborne radioactivity. In all cases, health physics-trained personnel shall evaluate the radiological conditions and specify the required items of protective clothing to be worn.

Respiratory protective devices are required in any situation arising from plant operations in which an airborne radioactive area exists or is expected. In such cases, the airborne concentrations are monitored by health physics trained personnel and the necessary protective devices specified according to concentration and type of airborne contaminants present.

Respiratory devices which may be available for use include:

a) Full-face air purifying respirator

b) Air supplied full-face respirators or hoods

c) Self-contained breathing apparatus.

Respirators are maintained by checking for mechanical defects, contamination, and cleanliness by qualified health physics personnel.

12.5.2.2 Radiation Instrumentation

Facilities are provided for the health physics-chemistry group. These facilities include both laboratory and health physics work areas. These facilities are equipped to analyze routine air samples and contamination swipe surveys. These facilities also serve as a central location for portable radiation survey instruments, respiratory protection equipment, and contamination control supplies.

Survey instruments are calibrated periodically, and maintenance records are provided for each instrument.

12.5.2.3 Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from the interpretation of the thermoluminescent dosimeter (TLD) badge. The direct reading pocket chamber provides day-by-day indication of external radiation exposure.

All persons subject to occupational radiation exposure are issued beta-gamma TLD badges and are required to wear such TLD badges at all times while within the Radiation Control Area.

Special or additional TLD badges are issued as may be required under unusual conditions. These devices are issued at the discretion of health physics personnel.

The TLD badges are processed on a routine basis at monthly intervals. The TLD badge of any individual is processed whenever it appears that an overexposure may have occurred.

A self-reading pocket chamber is issued in addition to a TLD badge to certain individuals whose work conditions make a day-to-day indication of exposure desirable. Pocket chambers are read, recorded, and re-zeroed regularly. Pocket chamber records furnish the exposure data for the administrative control of radiation exposure.

Neutron doses are determined by using survey data and stay times to calculate an exposure.

12.5.2.4 Facilities and Access Provisions

The plant site is divided into two categories, the Clean Area and the Radiation Control Area for radiation protection purposes as shown on Figure 12.5.2-1.

Entry to and exit from the Radiation Control Area is through the designated Access Control Point only (see Figure 12.5.2-2).

The Radiation Control Area shall be surveyed, classified, and conspicuously posted in accordance with 10CFR20 regulations and the Power Supply Group's Radiation Protection and Control Manual.

The general arrangement of the service facilities is designed to provide personnel decontamination and change areas (see Figure 12.5.2-2). A clean locker room is used to store individuals' clothing when they must dress out to perform work in the Radiation Control Area. A dress out area is provided in the Radiation Control Area. A supply of clean protective clothing for personnel is maintained in this area. Appropriate personnel contamination survey devices are located at the exit point of the Radiation Control Area so that personnel can survey themselves upon leaving the Radiation Control Area. A decontamination shower and washroom is located in the Radiation Control Area.

The spent fuel shipping cask decontamination facility has facilities to handle the decontamination of large items of equipment. This decontamination area contains a decontamination tank and service facilities.

A decontamination area is also provided within the machine shop for the decontamination of hand tools and small equipment.

CHAPTER 13
CONDUCT OF OPERATIONS

CHAPTER 13

CONDUCT OF OPERATIONS

LIST OF FIGURES

13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

13.1.1.1 Organizational Arrangements

Carolina Power & Light Company is engaged in the production, transmission, distribution, and sale of electric energy to residential, commercial, and industrial customers spread over a service area of 30,000 sq. mi. in North and South Carolina. The Company has extensive experience in the design, construction, startup, testing, operating, and staffing of modern generating facilities.

The corporate organization, which provides line responsibility for operation of the Company, is shown in Figure 13.1.1-1. Ultimate responsibility for operation of HBR rests with the Vice Chairman, reporting to the Chairman/President. The Vice Chairman assigns responsibilities to the CP&L Groups and Departments described below:

Power Supply and Engineering & Construction - The Executive Vice President, Power Supply and Engineering & Construction, reports to the Vice Chairman and is responsible for the planning, engineering, construction, operation, and maintenance of the generating, transmission, and associated facilities, and for the fuel and materials management of the generating facilities. There are three groups and two additional departments reporting to the Executive Vice President, Power Supply and Engineering & Construction: a) the Power Supply Group, b) the Fuel and Materials Management Group, c) the Engineering & Construction Group, d) the Corporate Nuclear Safety and Research Department, and e) the Corporate Quality Assurance Department (see Figure 13.1.1-2). The responsibilities of each of these groups and departments are described below:

a) Power Supply Group - The Senior Vice President, Power Supply Group, reports to the Executive Vice President, Power Supply and Engineering & Construction and is responsible for the operation and maintenance of the generating, transmission, and associated facilities. There are four departments within the Power Supply Group: 1) the System Operations Department, 2) the Nuclear Operations Department, 3) the Fossil Operations Department, and 4) the Technical Services Department (see Figure 13.1.1-3). The responsibilities of each of these departments are described below:

1) The System Operations Department is responsible for load dispatch and the operation and maintenance of transmission lines and substations (see Figure 13.1.1-4).

2) The Nuclear Operations Department operates and maintains the Company's nuclear generating facilities to produce and deliver electric power to the Company's transmission system. The department consists of three sections: (a) Brunswick Plant Section, (b) H. B. Robinson Plant Section, and (c) Harris Plant Section (see Figure 13.1.1-5).

(a) The Brunswick Plant Section is responsible for the operation, maintenance, and management of the Brunswick Plant.

(b) The H. B. Robinson Plant Section is responsible for the operation, maintenance, and management of the Robinson Plant.

(c) The Harris Plant Section is responsible for the startup, operation, maintenance, and management of the nuclear generating facilities at the Harris site.

3) <u>The Fossil Operations Department</u> operates and maintains the Company's fossil and hydro generating facilities. The department consists of four sections and three units: (a) Northern Area Fossil Section, (b) Central Area Fossil & Hydro Section, (c) Fossil Operations Administration Section, (d) Generation Operations & Maintenance Section, (e) Western Area Fossil and Hydro Unit, (f) Lee Plant Unit, and (g) Sutton Plant Unit (see Figure 13.1.1-6). While not directly involved in the operation of the Company's nuclear plants, this department represents a source of feedback for maintenance and operational problems common to all plants and is a reservoir of generating plant experience which could be applied to nuclear facility problems if required.

4. <u>The Technical Services Department</u> supports nuclear and fossil plant construction and operations. The department consists of five sections and one unit: (a) Licensing and Permits Section (b) Environmental Technology Section, (c) Lands Section, (d) Nuclear Training Section, (e) Radiological & Chemical Support Section, and (f) Emergency Preparedness Unit (see Figure 13.1.1-7).

(a) The Licensing and Permits Section acts as the Company's interface with the NRC's Office of Nuclear Reactor Regulation and Office of Inspection and Enforcement. This section coordinates the preparation and submittal of licensing documents, including Safety Analysis Reports, Environmental Reports, License Amendments, and responses to NRC concerns. The section is also responsible for any permits required by the Environmental Protection Agency and other state and federal agencies. In addition, this section conducts the Company's seismic monitoring and meteorological data collection programs.

(b) The Environmental Technology Section conducts the Company's environmental monitoring programs and plant analytical chemistry and metallurgical laboratory services.

(c) The Lands Section purchases properties for power plants.

(d) The Nuclear Training Section is responsible for training in the areas of operator (for both nuclear and fossil plants), technical, and craft training. This section is also responsible for the operation of the simulator and other facilities at the Harris Energy and Environmental Center. The primary purpose of the Nuclear Training Section is to assure that highly qualified

personnel are available to maintain and operate the Company's generating facilities in a safe and efficient manner. An onsite nuclear training unit is located at each of the Company's nuclear plants. The training supervisor at each nuclear plant is responsible for planning and conducting the operator training and retraining, as well as training and indoctrination programs for Company and contract employees. The training supervisor is responsible for coordinating craft-technical training at the Harris Energy & Environmental Center and vendor courses. The plant is responsible for the qualification and requalification of craft-technical employees.

(e) The Radiological & Chemical Support Section is responsible for providing staff support to the Nuclear Operations Department and Fossil Operations Department in the areas of health physics, chemistry, radiological, and environmental activities and for the effective operation of the environmental laboratory.

(f) The Emergency Preparedness Unit is responsible for the emergency preparedness support for CP&L nuclear plants. These responsibilities include formulating corporate emergency plans and procedures; interrelating with Federal, State, and Local agencies; coordinating plant emergency planning; and assuring that emergency exercises are conducted and documented.

Also reporting to the Senior Vice President, Power Supply Group is a Vice President, Special Projects. The Vice President, Special Projects is responsible for consultation and technical recommendations in support of operation and maintenance of operating plants and engineering and construction of new plant facilities. He accomplishes this with and through existing personnel and by utilizing plant expertise to effectively improve plant facilities and operation.

b) Fuel and Materials Management Group - The Senior Vice President, Fuel and Materials Management Group reports to the Executive Vice President, Power Supply and Engineering & Construction. He is responsible for management of the materials and fuel needs of the generating and transmission facilities. There are two departments in the Fuel and Materials Management Group: 1) the Fuel Department and 2) the Materials Management Department (see Figure 13.1.1-8). The responsibilities of each of these departments are described below:

1) <u>The Fuel Department</u> ensures the proper management of nuclear and fossil fuels used for the production of electrical power. The department is organized into three sections: (a) the Nuclear Fuel Section, (b) the Fossil Fuel Section, and (c) the Administration and Analysis Section (see Figure 13.1.1-9). The Nuclear Fuel Section is staffed with personnel having both the technical and managerial expertise required to ensure a timely and adequate supply of nuclear fuel, to provide necessary quality assurance for the fuel, to review fuel and core design, to support nuclear plant outages (including

refuelings) and operations, and to provide for spent fuel management. The Nuclear Fuel Section meets with members of the Company's operating nuclear plants on a continuing basis to plan and optimize the fuel operation strategy.

2) The Materials Management Department is responsible for Corporate, purchasing, inventory control, warehousing, and salvage of the Company's material needs (see Figure 13.1.1-10).

c) Engineering & Construction Group - The Senior Vice President, Engineering & Construction Group, reports to the Executive Vice President, Power Supply and Engineering & Construction and is responsible for the planning, engineering, and construction of generating, transmission, and associated facilities. There are five departments within the Engineering & Construction Group: 1) the Nuclear Plant Engineering Department, 2) the Fossil Plant Engineering & Construction Department, 3) the Nuclear Plant Construction Department 4) the Engineering & Construction Support Services Department, and 5) the Transmission and Communication Planning, Engineering and Construction Department (see Figure 13.1.1-11). The responsibilities of each of these departments are summarized below:

1) The Nuclear Plant Engineering Department provides engineering support for operating nuclear plants and engineering management for new nuclear construction projects. This department is divided into two major sections: (a) the Harris Plant Engineering Section and (b) the Engineering Support - Nuclear Plants Section (see Figure 13.1.1-12).

(a) The Harris Plant Engineering Section is responsible for providing the design and engineering for the Shearon Harris Nuclear Power Plant (SHNPP) project, including engineering support of site activities. This section is also responsible for accomplishing corporate, group, and departmental goals associated with the project. The section fulfills these responsibilities by managing the contract for architect/engineer services, providing technical direction for project design, and managing the procurement of engineered equipment. The primary responsibility of the Harris Plant Engineering Section is to provide a cost-effective, well-engineered power plant which optimizes operational considerations and allows efficient construction. The Harris Plant Engineering Section is located at the SHNPP site, and works on a day-to-day basis with the site construction organization, site start-up organization, and the SHNPP operations organizations.

(b) The Engineering Support - Nuclear Plants Section is responsible for providing engineering support for the Company's operating nuclear plants. The Engineering Support - Nuclear Power Plants Section is organized into the Brunswick Unit, the Mechanical Unit, and the Electrical Unit. The three units are responsible for providing the design engineering necessary to resolve operating plant problems referred to these units. Architect/Engineers may also be retained to assist the section in meeting its objectives.

The Director - Nuclear Engineering Safety Review is a position within the Staff Unit of the Nuclear Plant Engineering Department. He is responsible for engineering safety review of the department's activities and for processing operations and engineering feedback to prevent known problems from occurring in modifications to operating plants or in the design of the Harris Plant. In addition, he assures that a departmental training program is developed and implemented so that department personnel are trained in their responsibilities to meet NRC regulations, codes, standards, and other commitments made by the Company.

2) The Fossil Plant Engineering & Construction Department provides engineering and construction support and management for additions and modifications to operating fossil and hydro-generating plants and for new fossil generating plants. The department is divided into four sections: (a) Engineering Support, Fossil Plants Section I, (b) the Engineering Support, Fossil Plants Section II, (c) the Construction, Fossil Plants Section, and (d) the Special Projects Section (see Figure 13.1.1-13). While not primarily associated with the Company's nuclear generating facilities, this department represents a source for feedback of potential problems common to all types of plants and is also a reservoir of engineering and construction talent and experience which could be applied to problems at nuclear facilities if required.

3) The Nuclear Plant Construction Department manages the construction of nuclear generating facilities and is divided into three major sections: (a) the Brunswick and Robinson Site Management Section, (b) the Harris Site Management Section, and (c) the Construction Procurement and Contracting Section (see Figure 13.1.1-14).

(a) The Brunswick and Robinson Site Management Section is responsible for major additions and modifications and for construction work for CP&L's operating nuclear units. The section also supports the operating plants by performing or managing minor assigned tasks when requested by the Nuclear Operations Department.

(b) The Harris Site Management Section has the responsibility for construction management of the Harris project site and for the control over the constructor and contractors of the main plant site.

(c) The Construction Procurement and Contracting Section is responsible for the procurement of materials and equipment used during construction and modifications of generating facilities for the Company. The section also provides contracting services for Fossil Plant Engineering & Construction Department and Nuclear Construction Department construction projects.

4) The Engineering & Construction Support Services Department provides support services to other departments within the Engineering & Construction Group in the areas of estimating, schedule coordination, budgeting, cost control, cost reporting, construction accounting, and construction security (see Figure 13.1.1-15).

5) The Transmission & Communication Planning, Engineering and Construction Department is responsible for the planning, location, design, and construction of transmission facilities necessary to meet the bulk power requirements of the Company (see Figure 13.1.1-16). This Department also is responsible for the planning, design, and construction of Company-owned communications facilities.

d) Corporate Nuclear Safety and Research Department - The Vice President of the Corporate Nuclear Safety and Research Department reports to the Executive Vice President, Power Supply and Engineering & Construction. He is responsible for the management of the functions of corporate health physics, corporate nuclear safety, safety analysis review at the nuclear plants, and research in support of Company activities. The Corporate Nuclear Safety and Research Department is divided into three sections: 1) Corporate Health Physics Section, 2) Corporate Nuclear Safety Section, and 3) Research Section (see Figure 13.1.1-17). An onsite nuclear safety subunit at each nuclear facility reports independently offsite to the Corporate Nuclear Safety Section. This subunit was formed in an effort to stay in close contact with day-to-day operations and better access the manner in which plant activities are conducted.

e) Corporate Quality Assurance Department - The Manager of the Corporate Quality Assurance Department reports to the Executive Vice President, Power Supply and Engineering & Construction (see Figure 13.1.1-18). This department was organized to consolidate the quality assurance, quality control, and audit functions which were previously performed separately for engineering and construction, operations, and corporate quality assurance audit activities. In this manner, the Manager - Corporate Quality Assurance oversees the QA/QC activities of both the Power Supply and the Engineering & Construction Groups while maintaining independence from any responsibilities within those groups. The Corporate Quality Assurance Department is organized in four major divisions: 1) the Engineering & Construction QA/QC Section, 2) the Operations QA/QC Section, 3) the Performance Evaluation Unit, and 4) the Training and Procedures Unit. Their responsibilities are summarized below:

1) The Engineering & Construction QA/QC Section has the primary responsibility for the Harris Plant Quality Assurance/Quality Control in the engineering and construction phase and during start-up. Its purpose is to anticipate and preclude safety-related nonconformances. This section is also responsible for the preparation and distribution of the ASME "N" Stamp QA Manual.

2) The Operations QA/QC Section is responsible for assuring proper application of quality standards, practices, and procedures associated with plant operation, maintenance, or modification at the company's operating plants. An onsite QA/QC unit at each nuclear facility reports offsite to the Manager-Operations QA/QC Section in order to maintain the independence of their responsibilities. These units are responsible for conducting the site QA/QC activities in accordance with the Corporate QA Program and QA/QC procedures.

3) The Performance Evaluation Unit is responsible for conducting an independent corporate audit program. Both internal and external audits

are conducted. This unit assures by audit that all company functions related to nuclear power plants are carried out in accordance with the Corporate Quality Assurance Plan.

4) The Training and Procedures Unit is responsible for implementation of a training program designed to qualify QA/QC personnel for maximum interchangeability among various QA/QC activities; preparation and maintenance of the Corporate QA Manual, and QA/QC procedures for corporate and/or field use.

13.1.1.2 Qualifications

The CP&L Corporate Organization is fully qualified to support the operation of the Robinson Plant as documented by the issuance of the Facility Operating License.

13.1.2 OPERATING ORGANIZATION

13.1.2.1 Plant Organization

The facility organization is shown on Figure 13.1.2-1

13.1.2.2 Plant Personnel Functions, Responsibilities, and Authorities

General Manager - Robinson Plant

The Plant General Manager is responsible for all phases of plant management, including operation, maintenance, and technical supervision. He is responsible for adherence to all requirements of the Operating License and Technical Specifications. He is supported in these responsibilities by the Manager - Operations and Maintenance, Manager - Technical Support, Manager - Environmental and Radiation Control, Director - Planning & Scheduling, and the Assistant to General Manager. In the absence of the Plant General Manager, either the Manager - Operations and Maintenance, the Manager - Technical Support, the Manager - Environmental and Radiation Control, or the Director-Planning & Scheduling will assume his authorities and responsibilities. The Plant General Manager reports directly to the Vice President - Nuclear Operations.

Manager - Operations and Maintenance

The Manager - Operations and Maintenance is responsible for the operation and direct maintenance support of the unit, including refueling operations. His responsibilities are accomplished through those reporting to him, including the Maintenance Supervisor - Unit 2 and the Operating Supervisor - Unit 2. The Manager - Operations and Maintenance reports to the Plant General Manager.

Maintenance Supervisor - Unit 2

The Maintenance Supervisor - Unit 2 is responsible for ensuring that equipment, instrumentation, controls, and mechanical and electrical systems of Unit 2 are maintained at optimum dependability, safety, and operating efficiency in compliance with plant Technical Specifications, QA, Security, Radiation Control, and regulatory requirements. He accomplishes this by planning, directing, and controlling a highly skilled staff, inspecting maintenance work, providing effective maintenance procedures, standards, and developing improvements in the Preventive and Corrective Maintenance Program. He is assisted in these functions by foremen and engineers. The Maintenance Supervisor - Unit 2 reports to the Manager - Operations and Maintenance.

Operating Supervisor - Unit 2

The Operating Supervisor - Unit 2 is responsible for ensuring generation of the maximum amount of electric power from Unit 2 at optimum efficiency, reliability, and availability in compliance with plant Technical Specifications, QA, Security, Radiation Control, and regulatory requirements. His responsibilities are accomplished through those reporting to him, including the Principal Engineer - Operations and the Shift Foreman.

The Operating Supervisor - Unit 2 shall hold a Senior Operator's License. The Operating Supervisor - Unit 2 reports to the Manager - Operations and Maintenance.

Principal Engineer - Operations

The Principal Engineer - Operations is responsible for providing technical and engineering support to the operating group in the areas of safety analysis functions, plant operations, and fire protection to contribute to operating safety, efficiency, reliability, and compliance with regulatory requirements. This is accomplished through a staff including Shift Technical Advisors, Senior Engineer - Operations, and Senior Specialist - Fire Protection. The Principal Engineer - Operations reports to the Operating Supervisor - Unit 2.

Shift Technical Advisor

The Shift Technical Advisor is responsible for providing to the shift operations personnel technical advice (operating experience/accident assessment) dedicated to concern for the safety of the plant. He accomplishes this by performing engineering evaluations for plant operations, maintaining and broadening his knowledge of normal and off-normal operations, diagnosing off-normal events, and maintaining cognizance of current operating experience evaluations. The Shift Technical Advisor reports to the Principal Engineer - Operations.

Senior Engineer - Operations

The Senior Engineer - Operations is responsible for providing technical and engineering support to the operations group in the areas of plant, reactor, and radwaste operations to contribute to operating safety, efficiency, reliability, and compliance with regulatory requirements. The Senior Engineer - Operations reports to the Principal Engineer - Operations.

Senior Specialist - Fire Protection

The Senior Specialist - Fire Protection is responsible for ensuring effective plant fire protection and assisting in maintaining a safe and healthy work environment. This is accomplished by developing and implementing an effective plant Fire Protection Program which meets all specifications, regulations, and requirements. The Senior Specialist - Fire Protection reports to the Principal Engineer - Operations.

Shift Foreman

The Shift Foreman is responsible for ensuring the safe, dependable, and efficient operations of the Unit. He is responsible for adherence to all requirements of the appropriate items of the Operating License and Technical Specifications. As highest priority, at all times when on Control Room duty, it is the responsibility and authority of the Shift Foreman to maintain the broadest perspective of operational conditions affecting the safety of the plant. The Shift Foreman shall hold a Senior Operator's (SRO) license. The Shift Foreman, until properly relieved, shall remain in the Control Room at all times during accident situations to direct the activities of control room

operators. He may be relieved only by qualified persons holding SRO licenses. During routine operations when the Shift Foreman is temporarily absent from the Control Room, a Senior Control Operator shall be designated to assume the Control Room command function. The Shift Foreman is supported by and supervises Senior Control Operators, Control Operators, and Auxiliary Operators. The Shift Foreman reports to the Operating Supervisor - Unit 2.

Manager - Technical Support

The Manager - Technical Support is responsible for providing technical support for the efficient operation of the Robinson Plant in a manner to contribute to high plant efficiency, reliability, and availability. In addition, he is responsible for compliance with licenses, regulatory and Company requirements, nuclear safety, radiation control, and economic considerations. His responsibilities are accomplished through those reporting to him, including the Project Specialist - Regulatory Compliance and the Engineering Supervisor. The Manager - Technical Support reports to the Plant General Manager.

Project Specialist - Regulatory Compliance

The Project Specialist - Regulatory Compliance is responsible for providing regulatory support for plant efforts to comply with regulatory (NRC), NSAC, and Corporate Nuclear Safety requirements as well as INPO guidelines. This is accomplished by coordinating and following on-site NRC, CNS, QA/QC, INPO, and NSAC activities, inspections, commitments, responses, and by resolution of concerns. He ensures that commitments are met, responses which accurately depict the plant's position are submitted, reportable occurrences are detected and reported, documentation is maintained, and support for Technical Specifications and/or revision is provided. The Project Specialist - Regulatory Compliance reports to the Manager - Technical Support.

Engineering Supervisor

The Engineering Supervisor is responsible for all plant modifications and acceptance tests. He provides field technical direction and coordination for plant engineering studies. He is responsible for the development of the inservice inspection program. He develops procedures, instructions, and guidelines for plant engineering functions. He is supported in these tasks by a staff of Engineers and Engineering Technicians. The Engineering Supervisor reports to the Manager - Technical Support.

Director - Planning & Scheduling

The Director - Planning and Scheduling is responsible for planning, scheduling, and coordinating long- and short-term outages and backfit and modification work to provide the lowest downtime and/or least interference with plant operations and to maintain radiation exposures ALARA. He is supported in these activities by a staff of engineers. The Director - Planning & Scheduling reports to the Plant General Manager.

Senior Engineer - Planning & Scheduling

The Senior Engineer - Planning & Scheduling is responsible for developing and assisting in controlling plans, schedules, and coordination efforts for outages and backfit and modification work to support efforts to provide the lowest downtime and to maintain radiation exposures ALARA. The Senior Engineer - Planning & Scheduling reports to the Director - Planning & Scheduling.

Manager - Environmental & Radiation Control

The Manager - Environmental and Radiation Control is responsible for providing the environmental and radiation control support necessary for the operations of the plant within plant Technical Specifications and applicable state and federal regulations. His responsibilities are accomplished through those reporting to him including the Environmental & Chemistry Supervisor, the Radiation Control Supervisor, and Project Specialist - Radiation Control. The Manager - Environmental & Radiation Control reports to the Plant General Manager.

Environmental & Chemistry Supervisor

The Environmental & Chemistry Supervisor is responsible for planning, organizing, and directing chemical control and environmental surveillance programs. He is responsible for the appropriate laboratories, procedures, test results, and records. He is responsible for adherence to all requirements of the appropriate laboratories, procedures, test results, and records. He is responsible for adherence to all requirements of the appropriate items of the Operating License and Technical Specifications. He is assisted by a staff composed of a Project Specialist, various Foremen, and Technicians. He reports to the Manager - Environmental & Radiation Control.

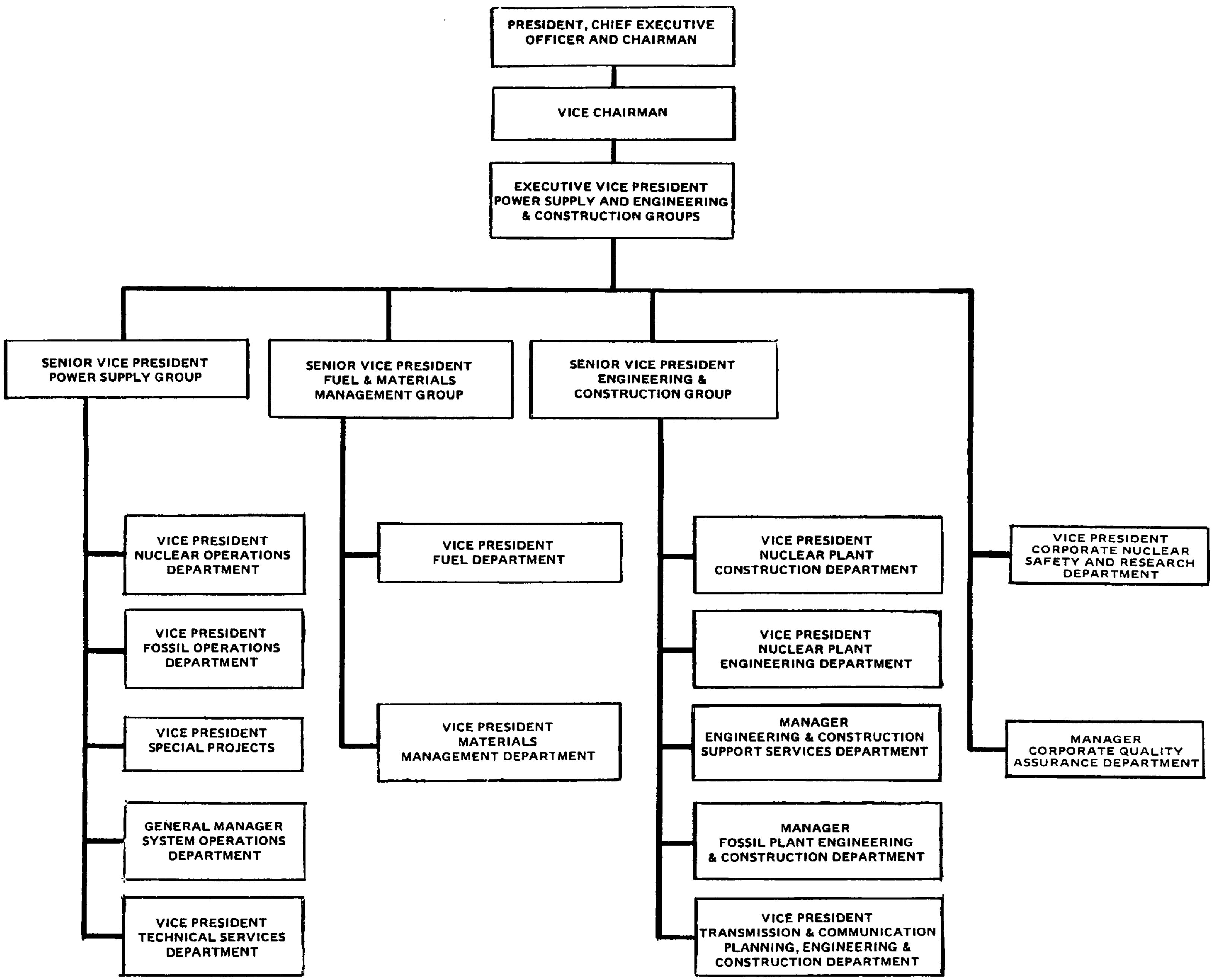
Radiation Control Supervisor

The Radiation Control Supervisor is responsible for providing the Radiation Control support necessary for the operation of the plant within plant Technical Specifications and applicable state and federal regulations. He is assisted by a staff of Radiation Control Foremen and Technicians. He reports to the Manager - Environmental & Radiation Control.

Project Specialist - Radiation Control

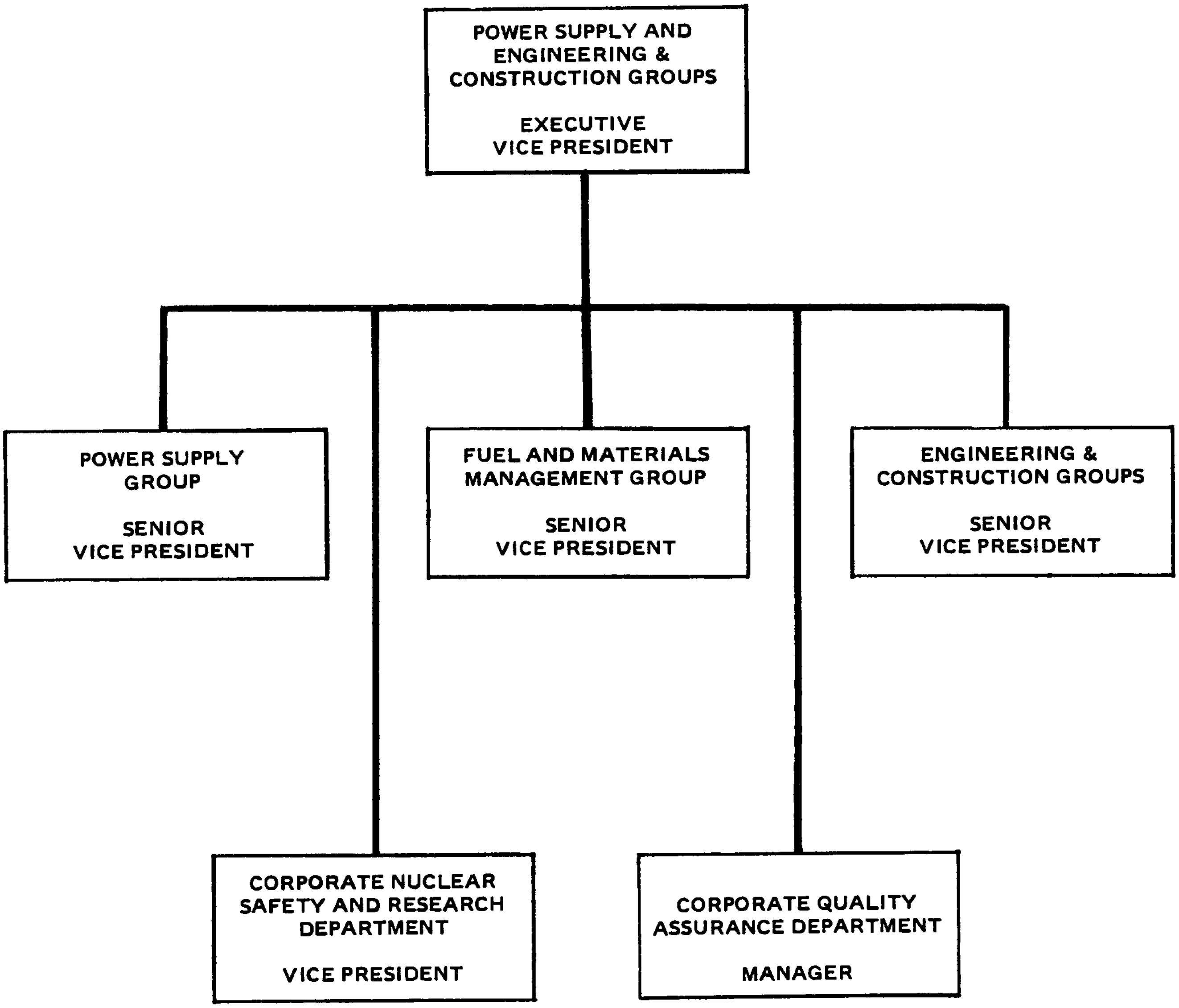
The Project Specialist - Radiation Control is responsible for providing technical and specialized support to the plant radiation control group to assist in efforts to operate the plant within Technical Specifications and applicable state and federal regulations. He is also responsible for ensuring that there is no adverse impact on the health and welfare of the public and employees, and for ensuring that radiation exposures are maintained as low as reasonably achievable (ALARA). He is assisted by a staff of Senior Specialists and Technicians. The Project Specialist - Radiation Control reports to the Manager - Environmental & Radiation Control.

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL
SAFETY ANALYSIS REPORT

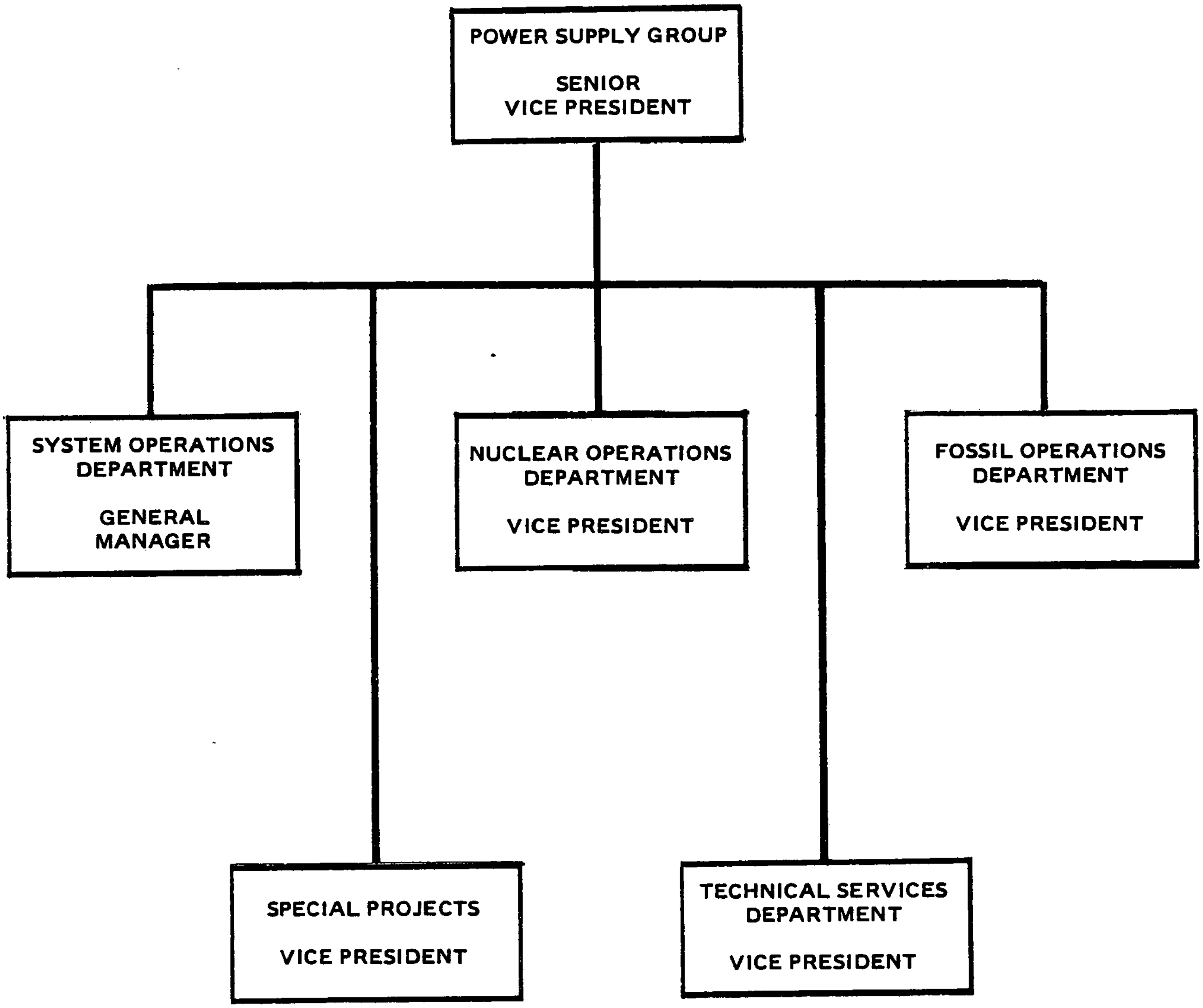


OPERATIONS AND ENGINEERING & CONSTRUCTION ORGANIZATION

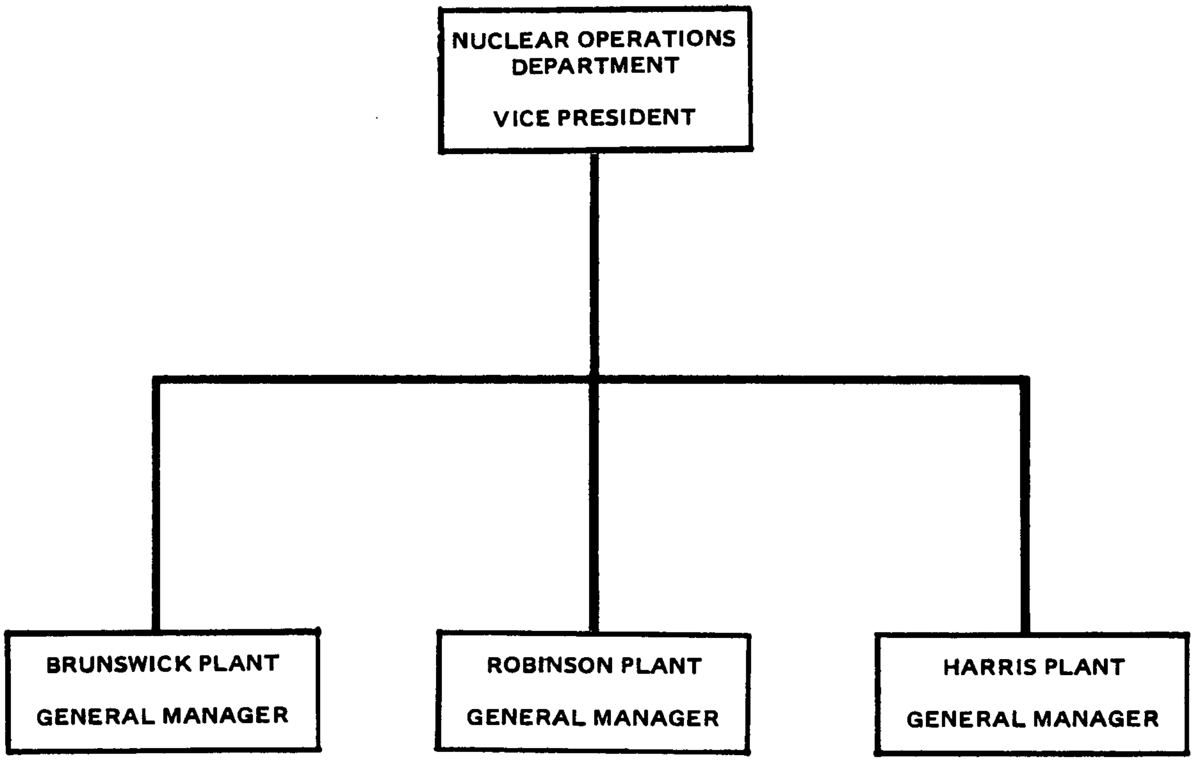
FIGURE 13.1.1-1



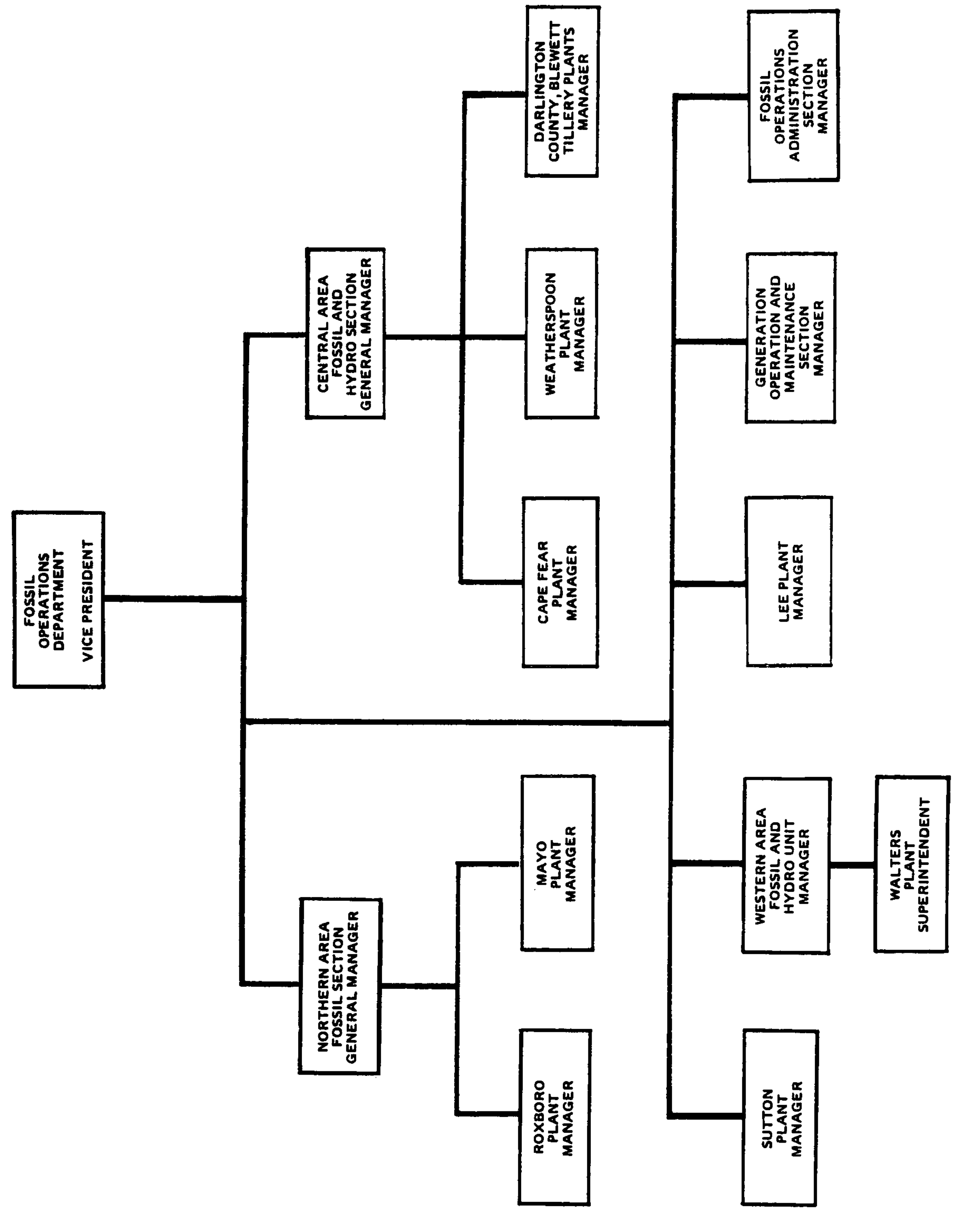
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	POWER SUPPLY AND ENGINEERING & CONSTRUCTION GROUPS ORGANIZATION	FIGURE 13.1.1 - 2



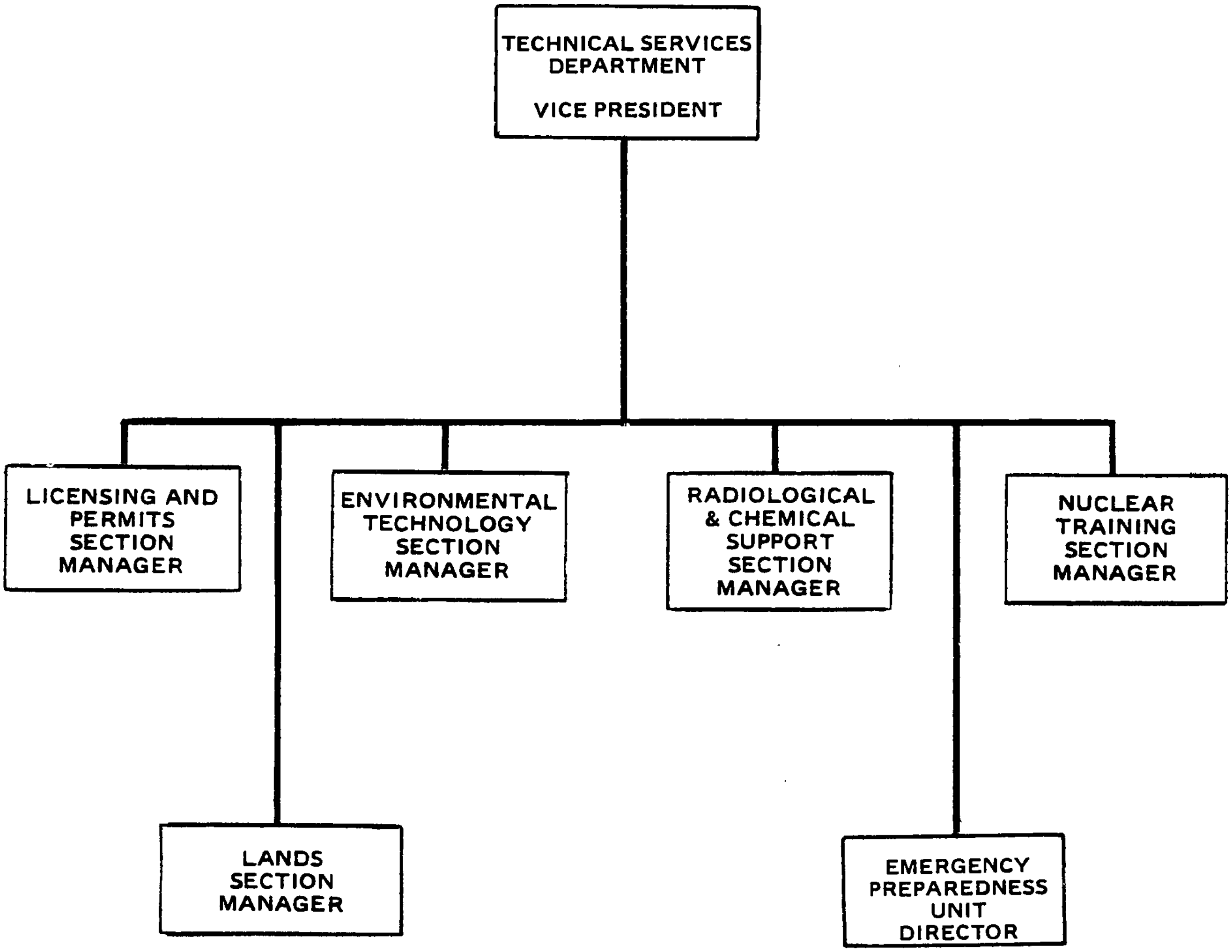
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	POWER SUPPLY GROUP ORGANIZATION	FIGURE 13.1.1 - 3



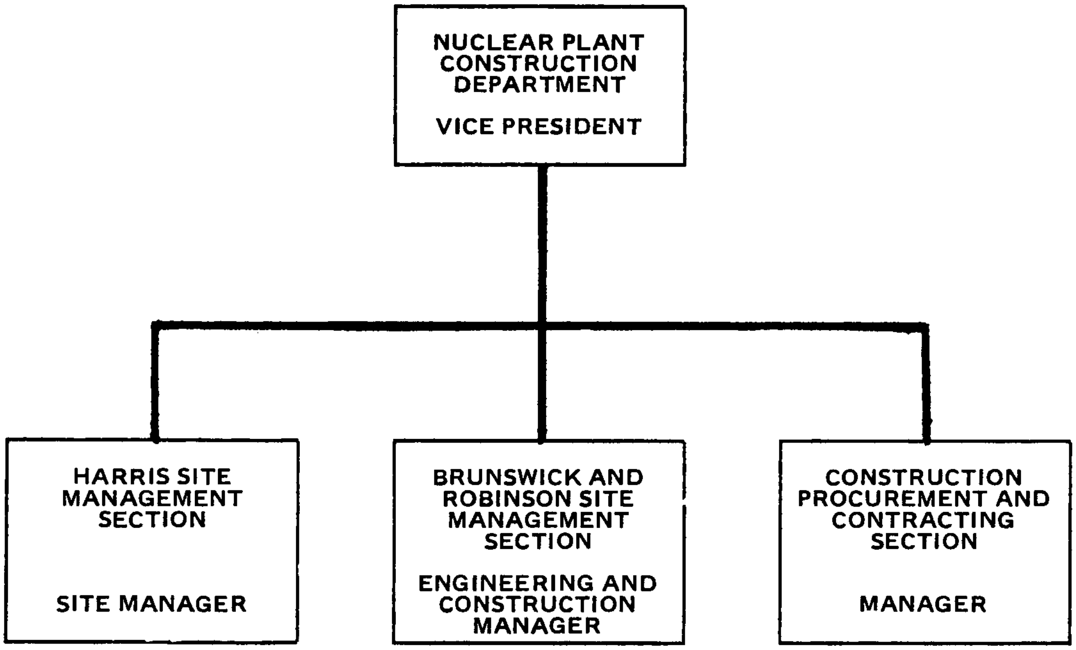
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	NUCLEAR OPERATIONS DEPARTMENT ORGANIZATION	FIGURE 13.1.1 - 5



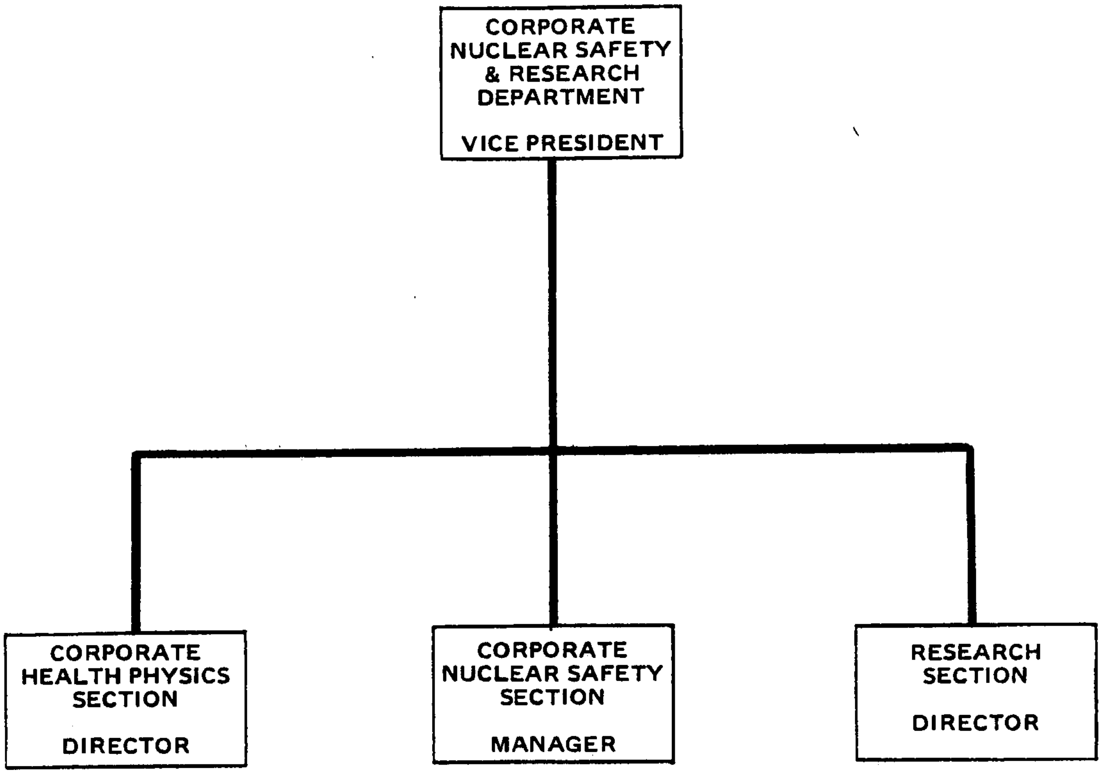
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	FOSSIL OPERATIONS DEPARTMENT ORGANIZATION	FIGURE 13.1.1 - 6



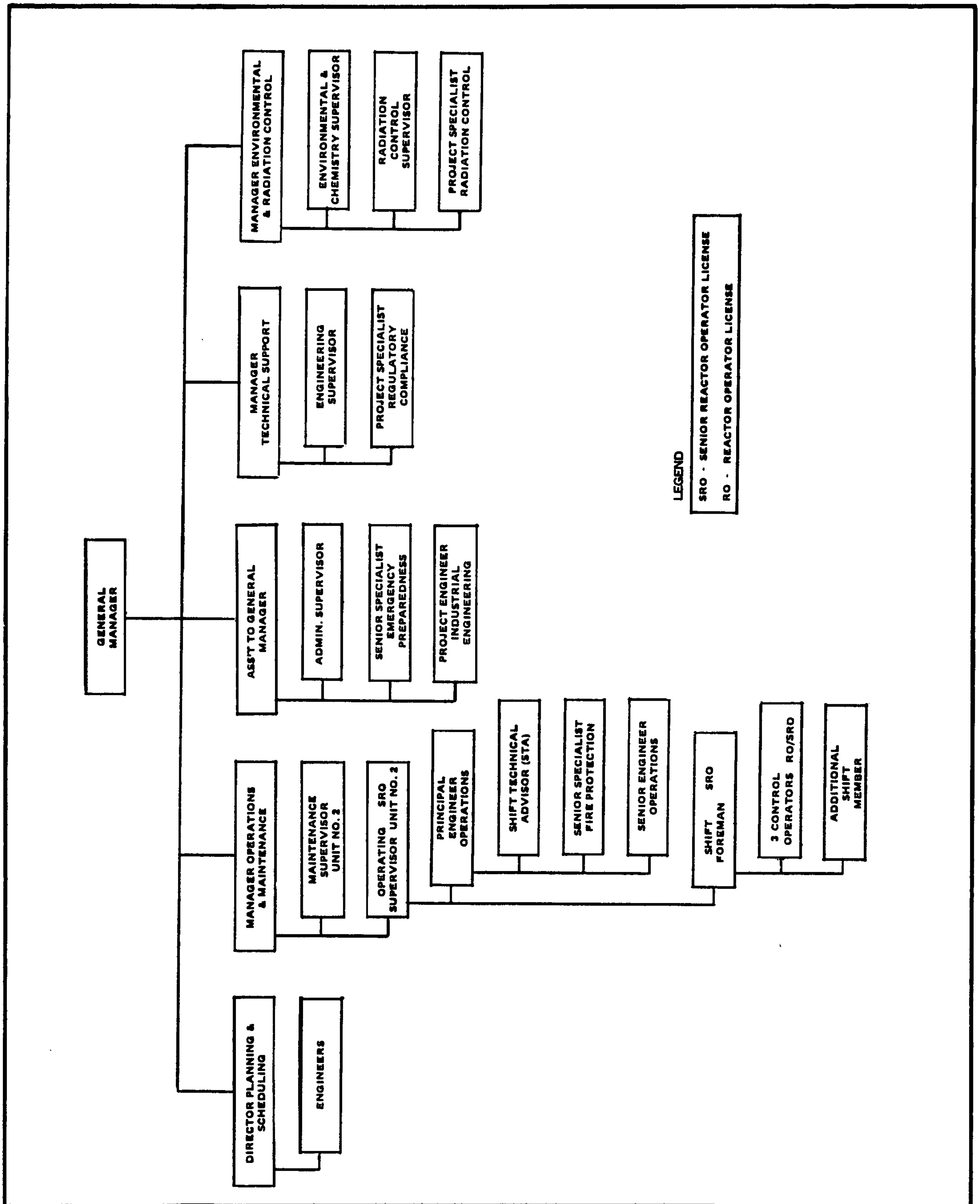
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	TECHNICAL SERVICES DEPARTMENT ORGANIZATION	FIGURE 13.1.1 - 7



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	NUCLEAR PLANT CONSTRUCTION DEPARTMENT ORGANIZATION	FIGURE 13.1.1 - 14



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CORPORATE NUCLEAR SAFETY & RESEARCH DEPARTMENT ORGANIZATION	FIGURE 13.1.1 - 17



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PLANT ORGANIZATION	FIGURE 13.1.2 - 1

13.3 EMERGENCY PLANNING

The description of plans for coping with emergencies at the H. B. Robinson Steam Electric Plant is contained in the latest revision of the H. B. Robinson Steam Electric Plant Radiological Emergency Response Plan, Volume 13 of the Plant Operating Manual (Reference 13.2.2-1).

CHAPTER 15
ACCIDENT ANALYSIS

LIST OF FIGURES

CHAPTER 15
ACCIDENT ANALYSIS

LIST OF FIGURES (Cont'd)

CHAPTER 15
ACCIDENT ANALYSIS

LIST OF FIGURES (Cont'd)

CHAPTER 15
ACCIDENT ANALYSIS

LIST OF FIGURES (Cont'd)

15.0 ACCIDENT ANALYSIS

This chapter addresses the representative initiating events listed in Regulatory Guide 1.70, Revision 3, as they apply to the H. B. Robinson (HBR) Nuclear Power Plant.

Certain items in the guide warrant comment, as follows:

There are no pressure regulators in the Nuclear Steam Supply System (NSSS) of a pressurized water reactor (PWR) design whose malfunction or failure could cause a steam flow transient.

No instrument lines from the Reactor Coolant System (RCS) boundary in the NSSS PWR design penetrate the Containment. For the definition of the RCS boundary, refer to ANSI-N 18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," Section 5, 1973.

Most analyses submitted in the original FSAR were updated when Exxon fuel was selected for core reloads. Where the revised analysis by Exxon was found to differ significantly from the original analysis (Reference 15.0-1), the original analysis has been either augmented or replaced.

Section 15.0.1 identifies the classification of all accidents evaluated in this chapter; this classification is consistent with the recommendation of Revision 3 to Regulatory Guide 1.70. A brief discussion of the core and coolant boundary protection analysis is provided in Section 15.0.2. Section 15.0.3 provides a summary of the plant characteristics and initial conditions assumed in the HBR safety analysis; the reactivity coefficients used are provided in Section 15.0.4. The rod cluster control assembly (RCCA) insertion characteristics, and trip set points and time delays assumed in the accident analysis are discussed in Sections 15.0.5 and 15.0.6, respectively. Finally, instrumentation drift and calorimetric errors used in assessing plant safety are provided in Section 15.0.7.

15.0.1 CLASSIFICATION OF ACCIDENTS EVALUATED

Consistent with the recommendations of Regulatory Guide 1.70, Revision 3, the accidents analyzed in this chapter of the FSAR are separated into the following categories:

a) Increase in heat removal by the secondary system (Section 15.1)

b) Decrease in heat removal by the secondary system (Section 15.2)

c) Decrease in RCS flowrate (Section 15.3)

d) Reactivity and power distribution anamolies (Section 15.4)

e) Increase in reactor coolant inventory (Section 15.5)

f) Decrease in reactor coolant inventory (Section 15.6), and

g) Radioactive release from a subsystem or component (Section 15.7).

The specific events evaluated in each section are discussed in that section.

A plant transient analysis of HBR 2 was performed to assess the impact on thermal margin of having 15 percent of the steam generator tubes plugged (Reference 15.0.1-1). The plant transients analyzed were those previously identified (Reference 15.0.1-2) to be the most severe with respect to thermal margin, i.e.:

a) Locked reactor coolant pump rotor at beginning of cycle (BOC)

b) Loss of reactor coolant flow at BOC

c) Slow rod withdrawal at BOC, and

d) Fast rod withdrawal at BOC.

The transients were analyzed with the first set of plant parameters listed in Table 15.0.3-1 and with neutronics parameters listed in Table 15.0.4-1. Temperatures, flows, and pressures in the primary coolant system remained the same as those used in previous analyses (Reference 15.0.1-3). Only the temperature and pressure of the secondary system were adjusted to account for a 15 percent reduction in steam generator tube heat transfer area.

The results of the analyses are summarized in Table 15.0.1-1. There is virtually no change in the mean departure from nucleate boiling ratio (MDNBR) due to steam generator tube plugging. The minimum MDNBR was 1.43 (including rod bow penalty) for the locked rotor transient. This result is well above the 1.30 limit.

Previous analyses (using the second set of plant parameters listed in Table 15.0.3-1) had considered additional plant transients (Reference 15.0.1-2); a review of these transients shows that they were not as limiting as those selected for analysis. The ones selected have shown only a small effect on thermal margin (less than 1 percent) due to plugging. These less severe incidents would not be expected to result in a worse MDNBR than those analyzed.

Based on the results of this analysis, it is concluded that, for up to 15 percent tube plugging, only a small change occurs in the MDNBR, with the resulting MDNBR still higher than the limiting MDNBR of 1.30.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from a pipe break, pump failures, valve malfunctions, or loss of outside AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, primary plant damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the core, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Loss of significant water from the RCS could conceivably lead to core damage.

The following provide the necessary protection against plant damage due to a loss of normal feedwater:

a) Reactor trip on very low water level in any steam generator

b) Reactor trip on steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator

c) Two motor-driven auxiliary feedwater pumps (300 gpm each), which are started automatically on the occurrence of conditions listed in Section 10.4.8.5, and

d) One turbine-driven auxiliary feedwater pump (600 gpm) which starts automatically on the occurrence of conditions listed in Section 10.4.8.5.

The motor-driven auxiliary feedwater pumps are supplied by the diesel-generator if a loss of offsite power occurs. The turbine-driven pump utilizes steam from the secondary systems. The turbine exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction directly from the condensate storage tank for delivery to the steam generators.

The above provides considerable backup in equipment and control logic to insure that reactor trip and automatic feedwater flow will occur following any loss of normal feedwater including that caused by a loss of AC power.

15.2.7.2 Analysis of Effects and Consequences

The analysis was performed to show that, following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat to prevent water relief through the pressurizer relief valves.

The following assumptions were made:

a) The initial steam generator water level (in all steam generators) at the time reactor trip occurs is at the very low level, this causes the reactor trip and automatic initiation of auxiliary feedwater flow. The initial water level is assumed to be at the lower narrow range level tap.

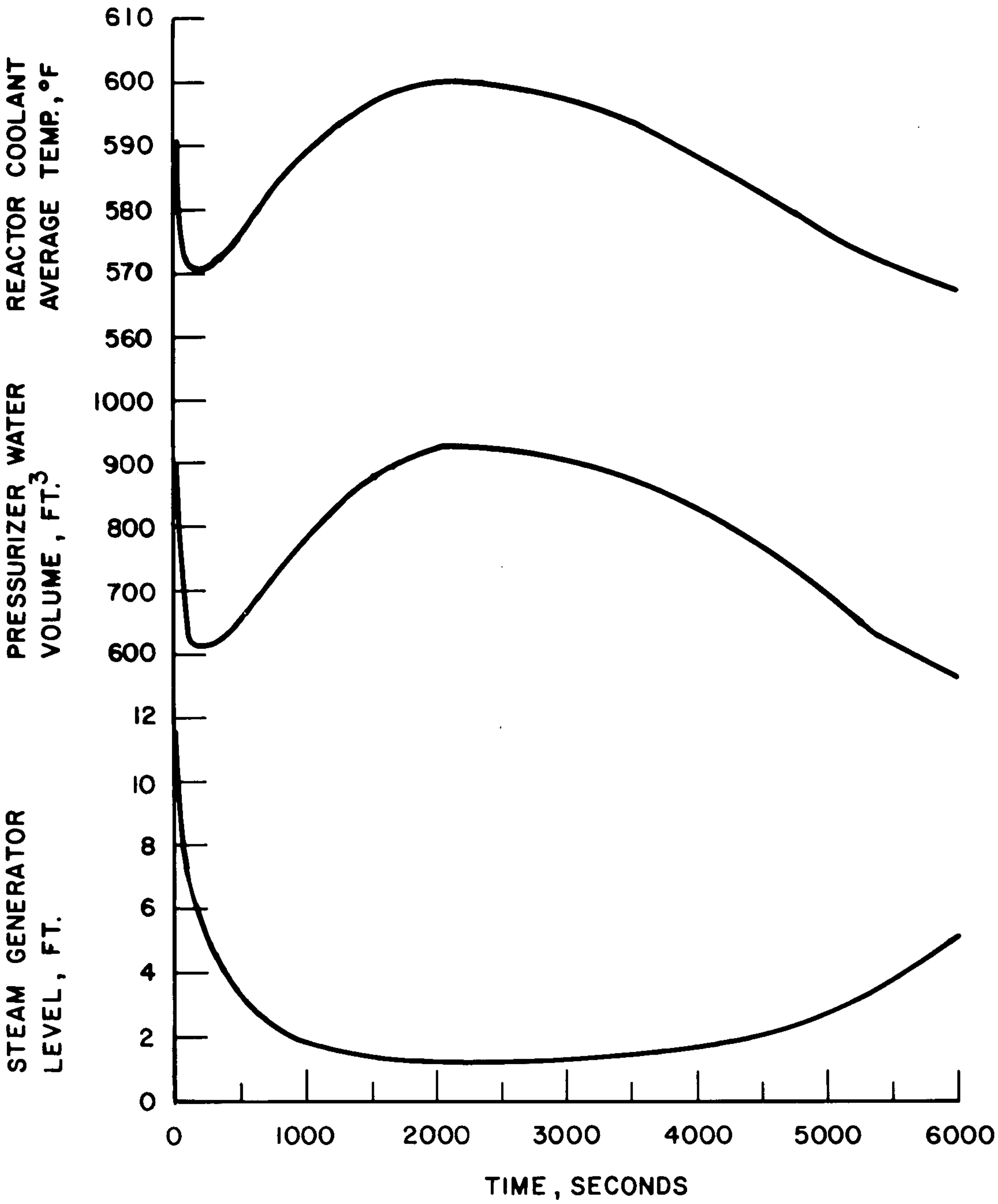
b) The plant is initially operating at 102 percent of 2300 Mwt, and

c) A heat transfer coefficient in the steam generators assuming RCS natural circulation.

The response to the transient for the RCS average temperature, the pressurizer water volume, and the steam generator water level is shown by Figure 15.2.7-1.

15.2.7.3 Conclusions

The loss of normal feedwater does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer relief or safety valves, nor does it result in uncovering the tube sheets of the steam generators being supplied with water.



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	TRANSIENT RESPONSE FOLLOWING A LOSS OF NORMAL FEEDWATER WITH ONE 300 gpm AUXILIARY FEED PUMP DELIVERING TO TWO STEAM GENERATORS BEGINNING AT ONE MINUTE	FIGURE 15.2.7 - 1

15.4.7 ROD CLUSTER CONTROL ASSEMBLY DROP

15.4.7.1 Identification of Causes and Accident Description

Dropping of a full length rod cluster control assembly (RCCA) could occur only when the drive mechanism is de-energized. This would result in a power reduction and an increase in the hot channel factor. If no protective action occurred, the RCS would restore the power to the level which existed before the incident. This would lead to a reduced safety margin or possibly DNB, depending upon the magnitude of the hot channel factor.

If a RCCA drops into the core during power operation (2244 Mwt), it would be detected by either a rod bottom signal device or by the use of the out of core chambers (Reference 15.4.3-1). The rod bottom signal device provides an individual position indication signal for each RCCA. Initiation of this signal is independent of lattice location, reactivity worth, or power distribution changes inherent with the dropped RCCA. The other independent indication of an RCCA drop is obtained by using the out of core power range channel signals. This rod drop detection circuit is actuated upon sensing a rapid decrease in local flux such as could occur from depression of flux in one region by a dropped RCCA. This detection circuit is designed such that normal load variations do not cause it to be actuated.

A rod drop signal from any rod position indication channel, or from one or more of the four power range channels, initiates protective action by reducing turbine load by a preset adjustable amount and blocking of further automatic rod withdrawal. Either action individually prevents core damage. The turbine runback is redundantly obtained by acting upon the turbine load limit and on the turbine governor control system. The rod stop is also redundantly actuated.

15.4.7.2 Analysis of Effects and Consequences

The transient following a dropped RCCA accident is determined by a detailed digital simulation of the plant. The dropped rod is assumed to cause a step decrease in reactivity and the core power generation is determined using a point neutron kinetics model. The overall plant response is calculated by simulating the turbine load runback and blocking of automatic rod withdrawal. The analysis is performed for the case in which the load cutback nearly matches the power decrease from the negative reactivity for a dropped rod (-2.0×10^{-3} δk), and also for the case in which the load cutback is greater than that required to match the worth of the dropped rod (-1.0×10^{-3} δk). In both cases the load is assumed to be cut back from 100 to 75 percent of full load at a conservatively slow rate of one percent per second. The actual amount of load cutback to be used will be determined during initial startup experiments and will be set to match the power reduction caused by the highest worth dropped rod.

The most negative values of moderator and Doppler temperature coefficients of reactivity are used in this analysis resulting in the highest heat flux during the transient. These are a moderator temperature coefficient of -3.5×10^{-4} $\delta k/^{o}F$ and a Doppler coefficient of -1.0×10^{-5} $\delta k/^{o}F$. A control group worth of 6×10^{-5} $\delta k/in.$ is assumed as equilibrium conditions are restored.

Figures 15.4.7-1 and 15.4.7-2 illustrate the transient response following a dropped rod of 2.0 x 10^{-3} δk at a power level of 2244 Mwt. The coolant average temperature decreases rapidly initially, then decreases slowly to new equilibrium condition. The peak heat flux following the initial response to the dropped rod is 95 percent of nominal. At the same time the core average temperature drops by $2^{o}F$ and the pressure by 28 psi.

Figures 15.4.7-3 and 15.4.7-4 illustrate the transient response following a dropped rod of 1. x 10^{-3} δk at 2244 Mwt. Again the coolant average temperature decreases initially, and then increases because of the negative reactivity feedback and the load cutback; The equilibrium temperature will again be achieved in about six minutes. For this case the peak heat flux following the initial response to the dropped rod is 96.5 percent of nominal. At the same time the core average temperature drops by $1^{o}F$ and the pressure by 40 psi.

An analysis has been made of the amount of flux tilt that can be tolerated without core damage for operating conditions of: 2244 Mwt power; core water inlet temperature of $550.2^{o}F$; primary pressure of 2220 psia. The effect of the flux tilt was represented by an increase in the radial heat flux hot channel factor. It was found that this factor could be increased by 12 percent before reaching a DNB ratio of 1.30. During initial startup experiments, it was verified that the flux tilt caused by the worst dropped rod, coupled with the thermal flux, coolant temperature, and primary system pressure responses will not result in a condition of DNB.

As shown by the startup physics tests (Reference 15.4.7-1) and the RCC assembly misalignment calculation (see Section 15.4.3 and Table 15.4.3-1) no inserted RCCA is capable of increasing core hot channel factors to the point at which DNB could occur, even assuming all operating conditions (power, temperature, and pressure) being at their most adverse values consistent with 2300 Mwt steady-state operation.

15.4.7.3 Conclusions

Protection for a dropped RCCA is provided by automatic turbine power cutback and blocking of automatic rod withdrawal. The magnitude of the power cutback is to be determined during the initial startup tests. As the analyses presented show, the protection system in conjunction with the load cutback, protects the core from DNB for a power tilt of 12 percent at maximum full power condition, greater than expected for the plant. At the reduced power condition following the rod drop, this allowable tilt will be even greater.

The power tilt will be experimentally determined and the protection system set to maintain a DNBR greater than 1.30.

An inserted or dropped RCCA does not jeopardize the core safety limits at either 2200 or 2300 Mwt. Therefore, automatic protective devices (such as turbine load reduction) are not required for safety.

Analysis of the RCCA drop with Exxon fuel in the core shows that safety margins are not reduced (Reference 15.0.1-2). The reactor does not return to power, and MDNBR does not fall below its initial value. Core average heat flux peaks at 89 percent of its initial value at about 18 sec. Core average

temperature drops only 3.3^{o}F with pressurizer pressure dropping 24.9 psia. No safety limits are exceeded, and the core approaches a new equilibriumcondition after 90 sec. The RCCA drop has an initial DNB ratio of 1.47. As the steam flow falls off, the reactor heat flux is reduced and DNB increases. The power level increases, causing the heat flux to start increasing again. The MDNBR reached during this heat flux increase is 1.77. Since steam flow continues to fall off, the reactor power decreases and new steady state conditions at about 75 percent power are approached.

For an RCCA drop, the power level is prevented from returning to full power by both an automatic turbine cutback and blockage of automatic rod withdrawal; this results in an immediate decrease in power level and a cool down. Therefore, a more positive moderator temperature coefficient will result in additional negative reactivity. For this reason, the transient was analyzed at end of cycle (EOC) conditions to provide a conservative result when the MTC is not negative. Since no change in the EOC value of the MTC is expected, a reanalysis is not necessary.

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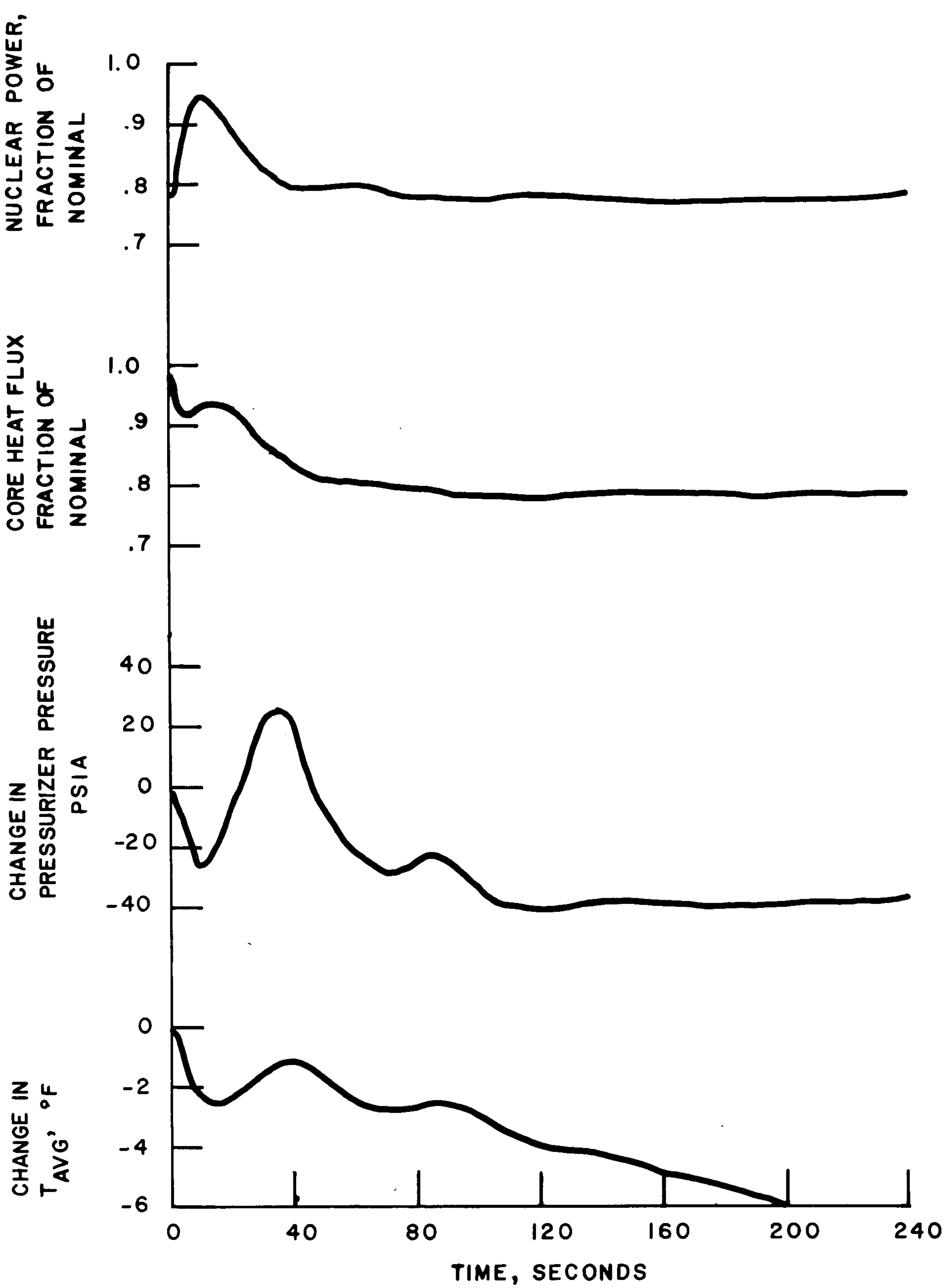

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	RESPONSES TO A DROPPED RCCA OF WORTH $2.0 \times 10^{-3}\ \delta k$ WITH A POWER CUTBACK OF 25% OF NOMINAL	FIGURE 15.4.7 - 1

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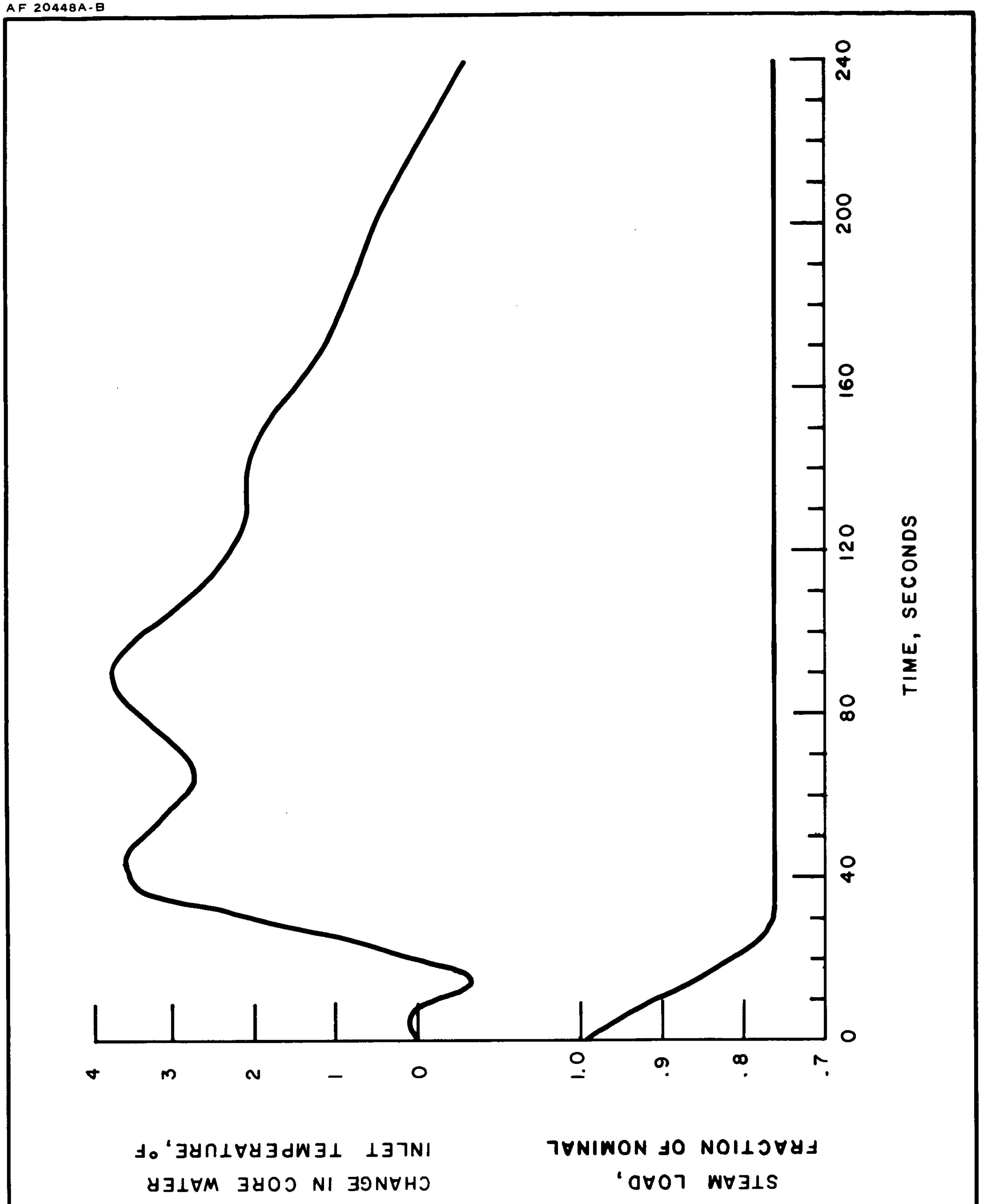

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	RESPONSES TO A DROPPED RCCA OF WORTH $2.0 \times 1.0^{-3}\ \delta k$ WITH A POWER CUTBACK OF 25% OF NOMINAL	FIGURE 15.4.7 - 2