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SUBJECT: Forwards Amend 12 to HB Robinson Steam Electric Plant,Unit 2 UFSAR.Amend include changes made under 10CFR50.59 not previously submitted.In future updated info transmitted by periodic submittals will be identified as "Revision."

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Carolina Power & Light Company
Robinson Nuclear Plant
PO Box 790
Hartsville SC 29550

May 13, 1994

10 CFR 50.71(e)
10 CFR 50.54(a)

Robinson File No.: 13510H
Serial: RNP/94-0999

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
SUBMITTAL OF UPDATED FINAL SAFETY ANALYSIS REPORT AMENDMENT 12

Gentlemen:

In accordance with the requirements of 10CFR50.71(e), Carolina Power & Light Company is submitting one original and ten copies of Amendment 12 to the Updated Final Safety Analysis Report (UFSAR) for the H. B. Robinson Steam Electric Plant, Unit No. 2. In accordance with the regulations, this amendment is submitted within 6 months following the most recent refueling outage.

Amendment 12 include changes made under the provisions of 10CFR50.59 but not previously submitted to the NRC. These changes have been appropriately located in the UFSAR. A description of these changes is not included in the submittal since a description is being submitted on the same date under a separate cover.

Amendment 12 presents changes made since the Amendment 11 submittal necessary to reflect information and analyses submitted to the NRC or prepared pursuant to NRC requirements. This Amendment is current through November 14, 1993.

In accordance with 10CFR50.54(a), this submittal includes changes to the Quality Assurance Program description which reduces a previously approved commitment and have been approved by the NRC. These proposed changes were submitted by letter dated March 23, 1993 (Serial No: NLS-93-086).

A053
1/11

As a Vice President of Carolina Power & Light Company, I certify that the information in this submittal accurately presents changes made since the previous submittal necessary to reflect information and analyses submitted to the NRC or prepared pursuant to NRC requirements.

In the future, the updated information transmitted by these periodic submittals will be identified as a "Revision" rather than an "Amendment" in accordance with the guidance contained in Generic Letter 81-06.

Questions regarding this matter may be referred to Mr. K. R. Jury at (803) 383-1363.

Very truly yours,

C S Hinnant

C. S. Hinnant
Vice President

JSK:dwm

Enclosure

c: Mr. S. D. Ebneter, Administrator, US Nuclear Regulatory Commission (NRC), Region II (w/one copy of Amendment 12)
Ms. B. L. Mozafari, US NRC Project Manager, HBRSEP
Mr. W. T. Orders, US NRC Senior Resident Inspector, HBRSEP

Carolina Power & Light Company
Robinson Nuclear Plant
PO Box 790
Hartsville SC 29550

May 13, 1994

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Mr. W. T. Orders, US NRC Senior Resident Inspector, HBRSEP

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CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

LIST OF TABLES

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SAR	Safety Analysis Report
SCR	silicon control rectifier
SFP	spent fuel pit
SHNPP	Shearon Harris Nuclear Power Plant
SG	steam generator
SI	safety injection
SIS	Safety Injection System
SOR	Senior Operator License
SRO	Senior Reactor Operator
SRWP	standing radiation work permit
SS	stainless steel
SSE	safe shutdown earthquake
SSPC	Steel Structure Painting Council
STP	standard temperature and pressure
SWP	service water pump
SWPS	Solid Waste Processing System

Regulatory Guide 1.88 — COLLECTION, STORAGE AND MAINTENANCE OF NUCLEAR POWER PLANT QA RECORDS (AUGUST, 1974)

ANSI Standard N45.2.9-1974 — COLLECTION, STORAGE, AND MAINTENANCE OF QA RECORDS

The requirements for collection, storage, and maintenance of QA records at HBR Unit 2 will be in accordance with ANSI N45.2.9-1974, subject to the following:

a) Section 5.4, Item 2 Loose Records: HBR complies with this requirement except for short periods of time during the microfilming process.

b) Section 5.6 states: "Permanent and temporary records storage facilities shall be so constructed or located as to protect contents from possible destruction by causes such as fire, flooding, tornadoes, insects, rodents, and from possible deterioration by a combination of extreme variations in temperature and humidity conditions."

QA records are stored in permanent and temporary facilities as follows:

1) One hour UL-rated fireproof file cabinets are utilized for temporary storage of records. These file cabinets are located at work stations throughout the plant and will contain the records until transmitted to the vault.

2) Permanent storage of QA records will be in the plant vault constructed to meet the requirements of this ANSI Standard. 5

3) Selected records may be stored off site by a QA Records Storage supplier, provided that supplier meets the applicable sections of this ANSI Standard.

c) ANSI N45.2.9, Section 5.6 states: "Structure, doors, frames, and hardware should be Class A fire-rated with a recommended four-hour minimum rating." The reinforced concrete vault structure has 18" thick walls and a 15" thick roof which encompasses the minimum thickness requirements of 8" for reinforced concrete as required by NFPA 232-1980. However, seals for penetrations through walls designed to provide fire protection in excess of three hours are not available. Doors and hardware are also not available having a four-hour rating and meeting the pressure differential from a Region I design basis tornado. Doors for the vault are designed and constructed to withstand the pressure differential of a Region I tornado and meet the requirements for a three-hour fire.

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CHAPTER 2
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CHAPTER 2
SITE CHARACTERISTICS

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2.1.3 POPULATION DISTRIBUTION

2.1.3.1 Population Within 10 Miles

The 1980 estimated resident population between zero and ten miles of the Robinson Plant is presented in Tables 2.1.3-1. Estimates were based on 1980 enumerated district maps, county highway maps which indicated locations of residences, and the preliminary 1980 U.S. Bureau of the Census data (Reference 2.1.3-1). On a large scale map concentric circles were drawn at 1, 2, 3, 4, 5, and 10 miles, with the center of the reactor as center point. Resident population within each of the resulting "standard nuclear display geographic units" was determined in either of two ways. For rural areas, the 1980 enumerated districts were superimposed on county highway maps. A ratio of "houses listed" for the 1980 enumerated districts to "houses shown" on the highway maps was calculated. The population of each standard nuclear display geographic unit was then determined by counting the houses on the highway map which fell into each standard nuclear display unit and adjusting the number by the ratio of "houses listed" to "houses shown". The adjusted number of houses was then multiplied by 3.2, a conservatively high estimate of population per household.

Within incorporated towns or in areas where houses were not indicated, the population within a standard nuclear display unit was determined by calculating the portion of the area of each enumeration district falling within the display unit. The population within each enumeration district was assumed to be evenly distributed throughout the area unless land use maps, road maps, or visual observations provided more accurate information of population distribution (Reference 2.1.3-2).

Table 2.1.3-1 also includes resident population projections for each Census decade through the projected plant life and the projected resident population for the year 1986. Projections were based on the 1980 resident population distribution and county growth patterns provided by the State of South Carolina (Reference 2.1.3-3). County growth patterns were assumed to apply evenly throughout each respective county.

Results showed that the area included parts of four counties: Darlington, Chesterfield, Kershaw, and Lee, and the total 1980 resident population was approximately 31,000. The majority of these residents live in or around the city of Hartsville, 3 miles SSE (7631 city/11,529 suburban). One other small concentration of resident population was indicated for the city of McBee, 7 miles NW (774 city). Other population within the area is generally considered to be rural.

2.1.3.2 Population Between 10 and 50 Miles

The 1980 estimated resident population between ten and fifty miles is presented in Table 2.1.3-2. Estimates were based on 1980 U.S. Bureau of the Census data (References 2.1.3-4, -5, -6 and -7) and were derived using the Electric Power Research Institute's Guidelines for Estimating and Forecasting Population Distributions Surrounding Reactor Sites (Reference 2.1.3-8). A large scale U.S. Bureau of the Census map was used, and concentric circles were drawn at distances of 10, 20, 30, 40, and 50 miles with the center of the reactor as center point. The circles were then divided into 22 1/2 degree

segments with each segment centered on one of the 16 compass points. Resident population within each of the resulting "standard nuclear display geographic units" were made using the smallest geographic units used by the U.S. Bureau of the Census. Where a Census unit did not fall entirely into a standard nuclear display geographic unit, population of such Census unit was distributed proportionally to the standard nuclear display geographic unit.

Table 2.1.3-2 also includes resident population projections for each Census decade through the projected plant life and the projected resident population for the year 1986. Methods used were similar to those described in Section 2.1.3.1 with the exception that projections for areas within North Carolina were based on county growth patterns provided by the State of North Carolina (Reference 2.1.3-9).

Results showed that the area is generally rural and is characterized by population concentrations in and around Florence, SC, 24 miles SE (30,062 city/9951 suburban); and Sumter, SC, 34 miles SSW (24,890 city/ 10,485 suburban). Cities with area populations over 10,000 include Laurenburg, NC, 44 miles ENE (11,480 city/536 suburban); Monroe, NC, 44 miles NNW (12,639 city/none suburban); Rockingham, NC, 42 miles NNE (8300 city/ 7203 suburban); and Lancaster, SC, 40 miles WNW (9603 city/6082 suburban).

2.1.3.3 Transient Population

The transient population within 10 miles of the Robinson Plant is composed of four major components: the industrial labor force, seasonal population variation, school population, and hospital/nursing home populations.

2.1.3.3.1 Industrial Labor Force

The 1980 total available work force in the 10-mile area surrounding the plant was approximately 12,400 (40% of total population) (Reference 2.1.3-10). The industrial work force (the largest single element of the work force) consisted of four principal components: the Hartsville area employees, west Hartsville employees, employees at the Robinson Plant, and McBee area employees. Distribution of industrial employees within the ten mile radius is indicated in Table 2.1.3-3. The largest segment of the industrial labor force is working during the period of 7 a.m. to 5 p.m., in or near Hartsville. During other times, this work force diminishes to less than one-fourth of the daytime industrial labor force, but is still located in the same area (Reference 2.1.3-2).

2.1.3.3.2 Seasonal Population Variations

Within the 10-mile area surrounding the plant there are no major seasonal population variations. During the entire year Lake Robinson is used for fishing, boating, picnicking, and other recreational activities. Based on a 1975 creek and recreational survey, the daily summer peak transient population is approximately 550-650 people (Reference 2.1.3-11). This figure would include people who are boating on Lake Robinson, as well as those using shore facilities. Also during the warmer months, Prestwood Lake, located on the north side of Hartsville, is utilized by local residents for recreation. Prestwood Lake is a comparatively small body of water, and it is estimated that 50-100 people would be using the area on a peak day.

Approximately 10 percent of the land area of the Sandhills State Forest and the Sandhills Wildlife Refuge Area fall within ten miles of the Robinson Plant. The primary users of the Sandhills State Forest are hunters, fishermen, hikers, and picnickers. The peak usage during summer weekends is about 150 people. However, most of these people are at Sugarloaf Mountain, which is outside of the 10-mile area (Reference 2.1.3-11).

The major tourist attractions and activities in the general surrounding area are the Southern 500 Race at Darlington Raceway (15 mi. ESE) and the Cotton Festival in Bishopville (14 mi. SSW). These are activities which occur only once a year and do not significantly affect the population within the ten mile radius.

2.1.3.3.3 Schools

Within the ten mile radius, during the 1980 school year, there were 7,890 students in attendance at the various public schools, private schools, and Coker College (Reference 2.1.3-2). The location of the school population is presented in Table 2.1.3-4.

2.1.3.3.4 Hospitals and Nursing Homes

Within the ten mile radius there is one hospital, one nursing home, and one center for handicapped children (Table 2.1.3-5). Byerly Hospital has 116 beds and is located in the sector that is 5 3/4 miles east-southeast of the Robinson Plant. The Saleeby Center houses approximately 45 incapacitated children, and is located in the same area as Byerly Hospital. The Morrel Convalescent Home houses approximately 132 senior citizens, and is located 7 1/2 miles east of the Robinson facility (Reference 2.1.3-2).

2.1.3.4 Low Population Zone

The "low population zone" is defined in 10 CFR 100.3(b) as "the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident." For the purpose of calculation, the low population distance was assumed to be 4.5 miles.

2.1.3.5 Population Centers

Figure 2.1.1-1 shows the location of population centers of over 25,000 people within a radius of 100 miles of the site. There are only two population centers as large as 25,000 within 50 miles of the plant: Florence, SC, 24 miles SE (30,062 city/9951 suburban) and Sumter, SC, 34 miles SSW (24,890 city/10,485 suburban).

TABLE 2.1.3-3

1981 LOCATION OF INDUSTRIAL WORK FORCE

22 1/2 DEGREE SECTOR	RANGE - MILES										
	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	TOTAL
N/A	400[1]										400
E					100[3]						100
ESE						3205[4]					3205
SE				750[6]							750
SSE								143[5]			143
NW								246[2]			246
TOTAL	400			750	100	3205		389			4844

NOTES: [1]Estimated labor force at Robinson Nuclear Power Plant: estimated peak labor for both Robinson 1 (fossil fired) during outage and Robinson 2 during normal operation.

[2]A. O. Smith Company (246); projected work force of 400 to 500 in two years.

[3]International Mineral & Chemical Corporation (100).

[4]Cokers Pedigreed Seed Company (220); Hartsville Manufacturing (390); Hartsville Oil Mill (125); Hartsville Mill, Div. Milliken (244); Red Fox Apparel (173); Sonoco Products (2053).

[5]Roller Bearing Corporation of South Carolina (143).

[6]L'Eggs Products (750).

References 2.1.3-2 (for locations); 2.2.2-1 (for numbers except number 3).

TABLE 2.1.3-4

1980 LOCATION OF SCHOOL POPULATION

22 1/2 DEGREE SECTOR	RANGE - MILES							
	3-4	4-5	5-6	6-7	7-8	8-9	9-10	TOTALS
ENE							184[1]	184
E		604[2]	132[3]					736
ESE		2642[4]	2484[5]					5126
SE			219[6]			190[7]		409
SSE	598[8]							598
S							55[9]	55
NW					782[10]			782
TOTALS	598	3246	2835		782	190	239	7890

NOTES: [1]Antioch Elementary (134), Byrd Town Academy (50)

[2]North Hartsville Elementary (604)

[3]Sonavista Elementary (132)

[4]Hartsville Jr. High School (791), Hartsville Sr. High School (1426), Carolina Elementary (425)

[5]Butler High School (439), Washington St. Elementary (523), First Baptist (296), Thornwell Elementary (464) Emanuel Baptist (296), Coker College (305), First Presbyterian (66), St. Mary's (12), Head Start Preschool (83)

[6]Southside Elementary (219)

[7]Thomas Hart Academy (190)

[8]Kelleytown Baptist Church (52), West Hartsville Elementary (546)

[9]Ebenezer School (55)

[10]McBee Elementary (364), McBee High School (418)

Reference 2.1.3-2

TABLE 2.1.3-5

LOCATION OF HOSPITALS AND NURSING HOMES

RANGE - MILES

SECTOR	4-5	5-6	6-7	7-8	TOTAL
E				132[1]	132
ESE		161[2]			161
TOTAL		161		132	293

[1]Morrel Convalescent Home

[2]Byerly Hospital (116 patients), Saleeby Center (45 children)

Reference 2.1.3-2 (Note: Byerly Hospital and Saleeby Center location corrected for this report)

REFERENCES: SECTION 2.1

2.1.3-1 U.S. Department of Commerce, Bureau of the Census.

2.1.3-2 Carolina Power & Light Company and NUTEC, Evacuation Time Estimates H. B. Robinson Steam Electric Plant, June 1981.

2.1.3-3 SC Division of Research and Statistical Services, "Preliminary Population Projections," Columbia, SC, 1982.

2.1.3-4 U.S. Department of Commerce, Bureau of the Census, "1980 Census of Population and Housing, North Carolina, Final Population and Housing Counts," PHC80-V-35, March 1981.

2.1.3-5 U. S. Department of Commerce, Bureau of the Census, "1980 Census of Population and Housing, South Carolina, Final Population and Housing Counts," PHC80-V-42, March 1981.

2.1.3-6 NC Office of State Budget," 1980 Population of Final Census Designated Places," Raleigh, NC, August, 1981.

2.1.3-7 SC Division of Research and Statistical Services, "Census Designated Places Profile," Columbia, SC.

2.1.3-8 Electric Power Research Institute, "Guidelines for Estimating and Forecasting Population Distributions Surrounding Reactor Sites," prepared by Sigma Research, Inc., EPRI EA-427-SR, 1976.

2.1.3-9 NC Office of State Budget and Planning, Research and Planning Services, "Update North Carolina Population Projections," Raleigh, NC, July 1981.

2.1.3-10 U.S. Department of Commerce, Bureau of the Census, "City and County Data Book," Washington, DC, 1973.

2.1.3-11 Testimony of Ralph L. Sanders and William T. Hogarth, EPA Public Hearing, Hartsville, SC, February 8, 1977.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 LOCATIONS AND ROUTES

The Robinson Plant is located in an area which is generally rural and undeveloped. Immediately west of the plant and along the eastern shore of Lake Robinson residential development has occurred, as well as the establishment of various public and private recreational areas. Other residential development within the 5-mile area surrounding the plant is confined to the Hartsville area (3 miles SSE).

A coal fired electric generating plant is located west and adjacent to the nuclear unit. The Darlington IC Plant is located approximately 1 1/3 miles NNW of the nuclear plant. Immediately north of the Darlington IC Plant there is a gas pipeline. Other industrial development within 5 miles is limited to the areas in and surrounding Hartsville (3 miles SSE).

Agricultural development has occurred within the five-mile area, especially in areas north and west of the plant.

Principal transportation routes or facilities include highways, a railroad line (1600 ft W), and a small airport (2 1/2 miles E). 4

There are no military bases within the five mile area.

2.2.2 DESCRIPTIONS

Residential development along the shores of Lake Robinson is confined to the eastern and northern shore of the lake. Since 1960 numerous permanent and vacation homes have been built above the 230 ft contour. Below the 230 ft contour, property owners have constructed small private piers, boat docks, and ramps; access is provided by lease agreements between landowners and CP&L.

Public recreational areas include Easterling's Landing, 1.7 miles NNE (a beach, picnic and paved boat launch area); Atkinson's Landing, 1.2 miles NNE (a beach and boat launch area); and J & M Marina 4300 ft E (a beach, paved boat launch, and boat gasoline facility). A small private sailboat club is also located on the lake, 2 miles NNE. All facilities are on the eastern lake shore. Several other areas provide recreational access to the lake, but this use is limited compared to that of other facilities.

East and adjacent to the nuclear unit (Unit 2), CP&L owns and operates a 185 Mwe coal fired electric generating plant (Unit 1). Unit 1 was placed in service in 1960.

The Darlington IC Plant (1 1/3 miles NNW) is a 572 Mwe internal combustion (oil) electric generating plant. The plant is owned by Westinghouse, Inc., but leased and operated by CP&L.

Carolina Pipeline transports natural gas via an underground pipeline (2 miles N). The pipeline transects Lake Robinson in an east/west direction. That part of the pipeline which crosses the discharge canal extends above ground.

Other industrial development within 5 miles of the plant is not extensive, and includes eleven firms which employ more than 100 people (Table 2.1.3-3). Principal products are paper products, textiles, fertilizer, seeds, and bearings (Reference 2.2.2-1). All of these firms are located in or near Hartsville (3 miles SSE).

Agricultural development has occurred within the five-mile area especially in areas north and west of the plant. Acreage to the north includes numerous peach orchards. Associated with the peach orchards is a fruit processing firm which processes and distributes local peaches, as well as other non-local produce.

Principal transportation routes include SC 151 (1/2 mile E), a 2 lane highway running north and south connecting McBee and Hartsville; and numerous state maintained secondary roads.

A small private airport is located 2 1/2 miles east of the plant. Only small aircraft use the runway.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAM

2.3.3.1 Onsite Operational Program

Collections of HBR onsite meteorological data began in April, 1974. A guyed, openlatticed tower supports the lower and upper levels of instrumentation. Wind direction, wind speed, and wind variance (sigma theta) are recorded at both levels. Ambient and dewpoint temperatures are measured at the lower level. The differential temperature between the upper and lower levels is measured by twin, redundant delta temperature systems operating simultaneously. Solar radiation and precipitation are collected near ground level. The wind sensors are mounted on 12-foot booms oriented perpendicular to the general NE-SW prevailing wind flow to minimize tower shadow effects. The temperature probes and lithium chloride dewpoint sensor are housed in Climet aspirated shields mounted on 8-foot booms. A complete specification of major system component operating conditions is presented in Table 2.3.3-1; component manufacturer and manufacturer model numbers may be found in Table 2.3.3-2. Operational sensor elevations are displayed in Table 2.3.3-3. Component sensor accuracies are outlined in Table 2.3.3-4.

The meteorological tower is located about 0.53 miles north of the Containment Building. The base of the tower is at the plant grade level of about 225 feet above mean sea level.

An environmentally controlled shelter, which houses recording instruments, signal conditioning devices, and remote data access equipment, is located near the tower, perpendicular to the prevailing wind flow to minimize air trajectory deviations.

The Westinghouse Environmental Monitoring System is the primary data collection system. This system converts sensor outputs to a proportional number of discrete pulses that are electronically integrated and recorded on magnetic tape in 15-minute averaging periods. Also, a direct readout of any parameter is possible with this system. A test jack for each parameter is provided so that a pulse test counter may be plugged into it. The counter sums the pulses produced in a specific time interval, and the subsequent pulse total can then be converted to engineering units by use of a formula of the form $y = mx + b$.

Esterline Angus Twin Strip Chart Recorders are used for providing an analog record of both the upper and lower level wind direction and speed to back up the Westinghouse System. In addition, 15 minute averaged upper and lower level wind speed and direction, both differential temperatures, and ambient temperature parameters are telemetered to the CP&L Raleigh General Office on an hourly basis via voice grade telephone lines to the site, giving us the capabilities of remotely detecting malfunctions of these parameters.

A micro-computer based sensor-system has also been installed at the HBR 2 meteorological site. This micro-computer (ADAC Model #1200) manufactured by ADAC Corporation, is based upon the Digital Equipment Corporation (DEC) LSI-11/23 micro-computer system. This system was selected because of its proven reliability in remote operation in the food processing and steel industries whose operational environments place a high demand upon the system integrity. Because the system was being adapted to collect electronic signals representing meteorological parameters, the software for the system was developed internally by the CP&L meteorological staff.

The ADAC system software scans each meteorological sensor input, except precipitation, once every ten (10) seconds. The precipitation input is scanned for a contact closure once per second. Each contact closure represents .01 in. of precipitation. These are then summed for a 15-minute total precipitation value. All other 10-second scan values are summed for a 15-minute period, then the average value for each meteorological parameter is obtained by performing a 15-minute mathematical average. If during a 15-minute averaging period, more than 33% of the 10-second scan values (30 individual scans) are not valid, the entire 15-minute averaged interval is then indicated to be unavailable (i.e., set to 9999.00).

The 15-minute averaged values for each parameter are stored internally within the CMOS memory of the ADAC system. The CMOS memory has the capacity to store up to four days of historical 15-minute averaged data. All CMOS memory is battery backed-up to prevent the loss of any stored data during brief power outages. The internal clock is completely battery operated. Thus, once time has been set, the clock remains running even during periods the system may be off or during brief power outages. Each 15-minute average data interval is marked with the current date and time, so that the "date/time" stamp on each recorded 15-minute averaged interval represents the ending time of the 15-minute average.

The ADAC hardware and software configuration allows up to three (3) remote locations to access this meteorological data acquisition system simultaneously. Each access port can display either the 15-minute averaged data for the most current period, a user specified previous period (up to four days previously), or the current 10-second scans of the meteorological sensors. The multitasking nature of the ADAC allows all of these actions to be accomplished without interruption of the sensor scanning or mathematical averaging of the data.

The HBR 2 ERFIS computer system accesses the ADAC meteorological sensor-system every 15-minutes to acquire the latest 15-minute averaged data. This information is stored in the ERFIS system and displayed in the Control Room on demand from the ERFIS terminal.

The ADAC meteorological sensor-system was developed by the CP&L meteorological staff and placed in operational service during 1987. To assure that the methodology employed by the ADAC system provides information which is consistent with that collected by the Westinghouse sensor-system, both meteorological data collection systems will be operated simultaneously for at least 12 to 18 months to provide conclusive proof that major differences do not exist between the systems. Until this comparison is completed, the Westinghouse sensor-system data will continue to serve as the historical meteorological record for the HBR 2 site.

2.3.3.2 Data Reduction

The Westinghouse sensor-system magnetic tape cassettes are changed and brought back to the general office for monthly processing. Computer programs convert all parameter pulse totals into engineering units. The data is then reviewed and checked for consistency with the onsite strip charts, Columbia, SC Weather Service data, and the onsite ADAC Sensor System. The edited 15 minute averaged data is then compiled into hourly averages and stored on magnetic

history tapes. Routine computer outputs include:

a) Monthly data summaries listing maximum temperature, minimum temperature, average temperature, barometric pressure, precipitation, solar radiation, and lower level dewpoint temperature as a daily average and monthly average.

b) Hourly averages of precipitation, barometric pressure, ambient temperature, differential temperature, lower level dewpoint, upper and lower level wind direction and wind speed, upper and lower level wind direction variance (sigma theta), Pasquill stability classes (as outlined in Regulatory Guide 1.23) computed from the average of the two delta temperature systems, and accumulated solar radiation (langleys/minute).

c) The 15-minute averages of both upper level and lower level wind direction, speed, and sigma theta, barometric pressure, and accumulated solar radiation.

d) Joint wind frequency distributions by direction (as outlined in Regulatory Guide 1.23) for both upper and lower levels, showing average wind speeds and number of unrecovered data hours.

The analog strip charts are changed twice per month. They are used as backup data to provide checks on the other systems and to provide consistency of data.

2.3.3.3 Maintenance and Calibration

An onsite maintenance and calibration program was initiated in January 1976. Regulatory Guide 1.23 data recovery requirements are met by performing scheduled calibrations carried out in accordance with Robinson Emergency Plan requirements such that:

a) All wind systems are changed and replaced with National Bureau of Standards traceable calibrated wind sensors, per Regulatory Guide 1.23.

b) All ambient and differential temperature systems are changed and replaced with NBS traceable calibrated systems, per Regulatory Guide 1.23.

c) The Lithium chloride dewpoint sensor bobbin is changed.

d) Calibrations of the barometric pressure, solar radiation, and precipitation systems are verified (sensors are changed on an annual basis).

e) All other onsite equipment is calibrated or its calibration is verified.

A further enhancement of data recovery is achieved by operating twin, redundant delta temperature systems simultaneously. Comparison of the two systems on a real time basis through the data (received at the CP&L General Office) gives us the capabilities to remotely detect discrepancies in either system, usually within 24 hours (except weekends).

2.3.3.4 Onsite Data

Westinghouse System onsite joint wind percentage frequency distributions (compiled per Regulatory Guide 1.23) for both upper and lower sensor elevations for the period January 1976 through December 1981 is presented in Tables 2.3.3-5 and 2.3.3-6. Data recovery percentages for this period are 98.1 percent for the lower level and 96.5 percent for the upper level.

All onsite joint wind frequency distributions were compiled by using the delta temperature stability classifications, as outlined by Regulatory Guide 1.23.

Average onsite windpseeds for the total six-year period at the lower and upper levels are 5.2 mph and 9.6 mph, respectively. Representation of the data to long term, area, climatological averages are discussed in Sections 2.3.1 and 2.3.2.

TABLE 2.3.3-1

OPERATING CONDITIONS

Wind Sensor:	-40°F to +120°F, up to 100 percent relative humidity, up to 125 mph wind speed
Temperature Sensors:	-50°F to +130°F
Aspirated Temperature Shields:	-60°F to +150°F
Honeywell Dew Point Sensor	-40°F to +160°F, 11 percent relative humidity and above
Total Precipitation Sensor:	No Limitations (equipped with heater)
Solar Radiation Sensor	No Limitations
Barometric Pressure Sensor:	-30°F to +170°F, 0 - 90 percent relative humidity
Magnetic Tape Recording Packages:	-20°F to +140°F
Strip Chart Recorder	+20°F to +120°F
Transmuter	-40°F to +120°F, 5 percent to 95 percent relative humidity
Telecoder R (Encoder):	0°F to +120°F, 0 to 100 percent relative humidity at +77°F to 104°F without condensation
Signal Conditioning Devices, Remote Data Access Equipment, and Data Acquisition Equipment:	+32°F to + 131°F

Table 2.3.3-2 (Con't)

COMPONENT	MANUFACTURER	MODEL NUMBER
Sigma azimuth amplifier circuit card	Meteorology Research, Inc.	14312
Power supply circuit card	Meteorology Research, Inc.	12784
Pulse transmitter for Honeywell dew point	Westinghouse	S1B4
Pulse transmitter for wind speed, wind direction (sine), wind direction (cosine), wind direction sigma, and barometric pressure	Westinghouse	S1B1
Pulse transmitter for solar radiation	Westinghouse	S1A1
Pulse transmitter for ambient temperature and differential temperature	Westinghouse	S1B3
Translator:		
Mainframe	Climatronics Corp.	F460
Temperature Translator Card	Climatronics Corp.	100869
Dew Point Translator Card	Climatronics Corp.	100870
Power Supply Card	Climatronics Corp.	F460
RECORDING DEVICES:		
Magnetic tape recorders	Westinghouse	WR-4C
Strip chart recorders for wind speed and direction	Easterline Angus	E1102R
REMOTE DATA ACCESS EQUIPMENT:		
Telecoder	Westinghouse	
Data phone set	Bell Telephone	103-A2

Table 2.3.3-2 (Con't)

COMPONENT	MANUFACTURER	MODEL NUMBER
REMOTE DATA ACCESS EQUIPMENT:		
Dialup modems	General Data Comm., Inc.	212A/SL
Dedicated 4-wire modem	General Data Comm., Inc.	202S/T
SUPPORT EQUIPMENT:		
Secondary power arrestor for input power lines	Dale	SPA-100
Isolation transformer	Topaz, Inc.	0111T25SR
Aspirated temperature shield for single-element temperature sensor	Climet	016-1
Aspirated temperature shield for dual-element temperature sensor and Honeywell dew point sensor	Climet	016-2
DATA ACQUISITION SYSTEM:		
Microcomputer Data Acquisition System	ADAC Corp.	System 1200
LSI-11/23 96Kb CMOS battery backup	ADAC Corp.	1816CMOS
A/D Converter 12 bit	ADAC Corp.	1012
Multiplexer Expander Card	ADAC Corp.	1012EX
Contact Closure Detector	ADAC Corp.	1616 CCI
Asynchronous Serial line Interface	ADAC Corp.	1750
Clock Card Battery Backup	Digital Pathways, Inc.	TU-50
CRT Display	Televideo	950

TABLE 2.3.3-4

COMPONENT ACCURACY

Wind Sensor:	
Wind Speed:	±0.4 mph or 1 percent, whichever is greater = 1.0 mph
Wind Direction, 0 to 540	±5.4 degrees
Honeywell Dew Point Sensor:	±2°F at or above 11 percent relative humidity
Solar Radiation Sensor: (pyranometer)	±0.04 calories/square centimeter/minute (langleys)
Differential Temperature System:	±0.51°F over ambient temperature range from -50 to +30°F
Ambient Temperature System:	±.31°F
Magnetic Tape Recorder:	±1 pulse per interval
Strip Chart Recorder:	±1 percent of full scale, Dir = ~ 5.4 degree, Speed = ~ 1.0 mph
Total Precipitation Sensor:	±0.5 percent (calibrated at 0.5 in. per hour)
Barometric Pressure Sensor:	±0.006 of mercury. (Temperature effect: ~0.1 in. of mercury per 100 degrees of Fahrenheit operating temperature span.)
Signal Conditions	±0.5% of full scale
A/D Converter:	±0.025% of full scale range
Amplifiers:	±.25% of full scale

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3.1.2.20 Protection Systems Redundancy and Independence

Criterion: Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure on removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. (GDC 20)

Response:

The RPS is designed so that the most probable modes of failure in each channel result in a signal calling for the protective trip. The protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious plant trip, or violate reactor protection criteria.

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. When the protective and control functions are combined, it is done only at the sensor. The protective and control functions are fully isolated in the remaining part of the channel, control being derived from the primary protection signal path through an isolation amplifier. Therefore, failure in the control circuit does not affect the protection channel.

The ESF equipment is actuated by one of the redundant ESF channels. Each coincident network energizes an engineered safety actuation device that operates the associated ESF equipment motor starters and valve operators. As an example, the control circuit for a SI pump is typical of the control circuit for a large pump operated from switchgear. The actuation relay, energized by the ESF Instrumentation System, has normally open contacts. These contacts energize the circuit breaker closing coil to start the pump when the control relay is energized. The ESF Instrumentation System actuates (depending on the severity of the condition) the SIS, containment isolation, and the Containment Air Recirculation Cooling System.

In the RPS, two reactor trip breakers are provided to interrupt power to the RCCA drive mechanisms. The breaker main contacts are connected in series (with the power supply) so that opening either breaker interrupts power to all full length RCCA, permitting them to fall by gravity into core.

Further detail on redundancy is provided through the descriptions of the respective systems covered by the various subsections in this section. Required continuous power supply for the protection systems is discussed in Section 8.3.1.

In summary, reactor protection is designed to meet all presently defined reactor protection criteria and is in accordance with the proposed Institute of Electrical and Electronic Engineers (IEEE) 279 "Standard for Nuclear Plant Protection Systems" August, 1968.

3.1.2.21 Single Failure Definition

Refer to Sections 3.1.2.20, 3.1.2.31, and 7.2.

3.1.2.22 Separation of Protection and Control Instrumentation

The physical arrangement of the redundant elements of the protection system are such that the probability is reduced that a single physical event will impair the vital function of the system (Section 3.1.2.23).

3.1.2.23 Protection Against Multiple Disability for Protection Systems

Criterion: The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis. (GDC 23)

Response:

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

The physical arrangement of all elements associated with the prospective system reduces the probability of a single physical event impairing the vital functions of the system.

Isolation of redundant analog channels originates at the process sensors and continues along the field wiring and through containment penetrations to the analog protection racks. Physical separation is used to the maximum practical extent to achieve isolation of redundant transmitters. Isolation of field wiring is achieved using separate wireways, cable trays, conduit runs, and containment penetrations for each redundant channel. Analog equipment is isolated by locating redundant components in different protection racks. Each channel is energized from a separate AC instrument bus.

System equipment is separated between instrument cabinets so as to reduce the probability of damage to the total system by some single event.

Wiring between vital elements of the system outside of equipment housing is routed and protected so as to maintain the true redundancy of the systems with respect to physical hazards.

3.1.2.24 Emergency Power for Protection Systems

Redundancy in emergency power is provided in that there are two DG sets capable of supplying separate 480 volt buses. One complete set of safety features equipment is therefore independently supplied from each DG. A third dedicated shutdown diesel is provided for redundant power for loads required for safe shutdown of the reactor.

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the initial control rods and soluble neutron absorber (boron).

3.1.2.29 Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn. (GDC 29)

Response:

The reactor core, together with the Reactor Control and Protection System is designed so that the minimum allowable DNBR is at least 1.17 and there is no fuel melting during normal operation including anticipated transients. | 3

The shutdown groups are provided to supplement the control group of RCCA to make the reactor at least one percent subcritical (K_{eff} = 0.99) following a trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCCA remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass, or relief valve. This is achieved with combination of control rods and automatic boron addition via the emergency core cooling system with the most reactive rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown.

3.1.2.30 Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public. (GDC 30)

Response:

The reactivity control systems provided are capable of making and holding the core subcritical under accident conditions in a timely fashion with appropriate margins for contingencies. Normal reactivity shutdown capability is provided within 2 sec following a trip signal by control rods with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

Boric acid is pumped from the boric acid tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps which inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of offsite power. Boric acid can be injected by one pump at a rate which takes the reactor to hot shutdown with no rods inserted in less than fifteen minutes. In fifteen additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level does not begin until approximately 15 hr after shutdown. If two boric acid pumps are available, these time periods are reduced. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water tank. This solution can be transferred directly by the charging pumps or alternately by the SI pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

3.1.2.31 Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31)

Response:

The RPS is designed to limit reactivity transients to DNBR $\geq$ the safety limit (specified in Section 4.4) due to any single malfunction in the deboration controls.

The RPS is capable of protecting against any single anticipated malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod and continuous deboration are described in Sections 15.4 and 9.3.4, respectively.

The release of fission products from the containment is limited in three ways:

a) Blocking the potential leakage paths from the containment. This is accomplished by:

1) A steel-lined, concrete reactor containment with continuously pressurized penetrations and testable liner weld channels

2) Isolation of process lines by the Containment Isolation System (CIS) which imposes double barriers in each line which penetrates the containment

3) An Isolation Valve Seal Water System which creates a leaktight seal between the valves in each line which penerates the containment water

b) Reducing the fission product concentration in the containment atmosphere by spraying chemically treated borated water which removes airborne elemental iodine vapor by washing action.

c) Reducing the containment pressure and thereby limiting the driving potential for fission product leakage by cooling the containment atmosphere using the following independent systems of essentially equal heat removal capacity:

1) Containment Spray System

2) Containment Air Recirculation Cooling System

3.1.2.38 Reliability and Testability of Engineered Safety Features

Criterion: All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public. (GDC 38)

Response:

A comprehensive program of plant testing is formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime.

The initial tests of individual components and the integrated test of the system as a whole, complement each other to assure performance of the system as designed and to prove proper operation of the actuation circuitry.

The engineered safety features components are designed to provide for routine periodic testing.

Plant Technical Specifications specify the test frequency and acceptance criteria to be used for periodic verification of the operability of engineered safety features actuation circuits and components.

The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupt the logic matrix output to the master relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the logic matrices to the master relay is complete is accomplished by use of an ohmmeter to check continuity. Additional verification is provided by periodically operating the safeguards pumps by means of their normal controls.

3.1.2.39 Emergency Power

Criterion: An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component. (GDC 39)

Response:

Independent alternate power systems are provided with adequate capacity and testability to supply the required engineered safety features and protection systems.

The plant is supplied with normal, standby and emergency power sources as follows:

a) The normal source of auxiliary power during plant operation is the generator. Power is supplied via the unit auxiliary transformer which is connected to the main leads of the generator.

b) Power required during plant startup, shutdown, and after reactor trip is supplied from the CP&L 115 kV system by a tap from the Robinson 115 kV switchyard to startup transformer No. 2.

c) Two diesel generator sets are connected to the emergency buses to supply power in the event of loss to all other AC auxiliary buses.

d) A dedicated shutdown system exists which will bring the plant to a safe shutdown condition in the event a fire or other event causes loss of all other power.

e) Emergency power supply for vital instruments, control, and some emergency lighting is supplied from two 125 V DC station batteries.

The DG sets are located in the Reactor Auxiliary Building and are connected to separate 480 V auxiliary system buses. Each set will be started automatically on a SI signal or upon under-voltage on its corresponding 480 V auxiliary bus. Each diesel is adequate to supply the engineered safety features for the hypothetical accident concurrent with loss of outside power. This capacity is adequate to provide a safe and orderly plant shutdown in the event of loss of

outside electrical power. The dedicated shutdown diesel is located in a separate enclosure and carries loads to bring the plant to a safe shutdown if all other power and control is lost including the DG (Section 8.3).

3.1.2.40 Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

Response:

The dynamic effects during blowdown following a LOCA are evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Fluid and mechanical driving forces are calculated, and consideration is given to possible damage due to fluid jets and secondary missiles which might be produced.

The steam generators are supported, guided, and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a RCS pipe rupture. These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

A LOCA or other plant equipment failure might result in dynamic effects or missiles. For such engineered safety features as are required to ensure safety in the event of such an accident or equipment failure, protection from these dynamic effects or missiles was considered in the layout of plant equipment and missile barriers. Fluid and mechanical driving forces were calculated, and consideration was given to the possibility of damage due to fluid jets and missiles which might be produced by the action jets. Consideration was given during the design to potential sources of missiles.

Layout and structural design specifically protect injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Individual injection lines penetrate the main missile barrier, and the injection headers are located in the missile-protected area between the missile barrier and the containment outside wall for the hot leg SIS and RHR system or outside containment for the cold leg SIS. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. Separation of the individual injection lines is provided to the maximum extent practicable. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment and which might be directed toward it so that no LOCA can result.

All hangers, stops, and anchors were designed in accordance with United States American Standards (USAS) B31.1 Code for Pressure Piping and ACI 318 Building Code Requirements for Reinforced Concrete which provide minimum requirements on material, design and fabrication with ample safety margin for both dead and dynamic loads over the life of the plant.

3.1.2.41 Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Response:

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

The extreme upper limits of public exposure are taken as the levels and time periods presently outlined in 10CFR100, i.e. 300 rem to the thyroid in two hours at the exclusion radius and 300 rem to the thyroid over the duration of the accident at the low population zone distance. The accident condition considered is the hypothetical case of a release of fission products per the Atomic Energy Commission's technical information report TID 14844. Also, the total loss of all outside power is assumed concurrently with this accident. However, operation of the SIS, considering the single failure criterion, limits the release of fission products from the core to only the gap activity between the fuel pellet and clad.

Under the above accident condition, the Containment Air Recirculation System and the CSS were designed and sized so that either system is able to supply the necessary post-accident cooling capacity to rapidly reduce the containment pressure following blowdown and cooling of the core by SI. The spray system was designed to provide adequate iodine removal with partial system effectiveness. Partial effectiveness is defined as operation of a system with one active component failure.

Each of the auxiliary cooling systems which serves an emergency function provides sufficient capability in the emergency operational mode to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the plant personnel and the public.

3.1.2.42 Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a LOCA to the extent of causing undue risk to the health and safety of the public. (GDC 42)

TABLE 3.2.1-1

SEISMIC CLASSIFICATION OF BUILDINGS AND STRUCTURES

STRUCTURE	CLASS
Containment (including all penetrations and air locks, the concrete shield, the liner and the interior structures)	I
Spent fuel pit	I
Control room	I
Diesel generator room	I
Intake structure (to the extent that water is always available to the service water pumps)	I
Auxiliary building	I
Turbine structure	II
Radwaste Facility	II
Buildings containing conventional facilities	III

TABLE 3.2.1-2 (Cont'd)

ITEM	CLASS	
Waste Disposal System All elements not listed as Class I	III	
Containment polar crane	I	
Manipulator and other cranes	III	
Conventional equipment, tanks and piping, other than I and II Classes	III	
AFW suction and discharge piping and pumps, Service Water and Fire Protection Systems* pumps and piping	I	7
The Chemical and Volume Control System (except the batching tank, monitor tank, monitor tank pumps, chemical mixing tank, and the resin fill tank)	I	
Batching tank	II	
Monitor tanks	II	
Monitor tank pumps	II	
Chemical mixing tank	II	
Resin fill tank	III	
Main steam piping from steam generator up to and including the first isolation valve	I	
Steam supply lines to the steam-driven AFW pump	I	7
Steam generator safety and relief valves	I	
Service water booster pumps and piping	I	
Spent fuel pit storage racks	I	
Primary water storage tank	I	
Containment vacuum and pressure relief systems	I	
Containment purge system	I	
Hydrogen Recombiner Supply and Return Piping	III	

*Parts as specified in Section 9.5.1 (only parts in Class I areas)

The redundancy and location of vital equipment is as follows:

(a) Emergency steam generator feed is provided by a steam-driven pump backed up by two motor-driven pumps. Both motor driven pumps are inside buildings. In the event the steam lines supplying steam to the turbine-driven pump are damaged, the motor driven pumps, powered by the emergency diesel generators located in Auxiliary Building, can be used.

(b) The four service water pumps are located in three separate bays in the intake structure, the middle bay containing two pumps. The pumps are sufficiently isolated to make it unlikely that a missile could damage more than one pump. The walls separating the bays and the deck above the piping are two and one half foot thick reinforced concrete. Thus it is highly unlikely that a missile could get to the pumps.

(c) The Condensate Storage Tank could be pierced by a missile. However, missile impact would have to occur at the bottom of the tank to cause total loss of water. Also, the service water system or the well water system could be used to supply water to the secondary side of the Steam Generators to remove decay heat.

(d) If a tornado or tornado debris destroys the outside electrical power supply, the unit could be tripped and either of the two emergency diesel generators, located in the Auxiliary Building, would supply sufficient power to place and maintain the plant in the safe shutdown condition. Also, a dedicated shutdown diesel is available for plant shutdown.

(e) Piping and electrical connections from the Auxiliary Building to the Containment are each in two separate concrete enclosures following different routes. Other vital piping and equipment is located in below-grade trenches or pits with concrete covers.

Therefore it is concluded that the plant can withstand the effects of the design tornado without endangering the health and safety of the public.

REFERENCES: SECTION 3.3

3.3.1-1 ASCE PAPER NO. 3269 "Wind Forces on Structures", Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

3.3.1-2 Letter, LAP-83-385, dated August 29, 1983, from S. R. Zimmerman (CP&L) to G. Requa (NRC), Auxiliary Feedwater System Evaluation.

3.3.1-3 Letter, NLS-85-168, dated March 14, 1985, from S. A. Varga (NRC) to E. E. Utley (CP&L), Feedwater System Tornado Missile Protection (TAC No. 49223). 4

3.3.1-4 Letter, NLS-85-208, dated June 13, 1985, from S. R. Zimmerman (CP&L) to S. A. Varga (NRC), Auxiliary Feedwater System Tornado Protection.

3.11.4 LOSS OF VENTILATION

Plant areas containing safety related equipment and their support systems are provided with temperature controlled environment during normal and worst case DBA conditions if required. The environmental parameters for different plant areas are presented in Section 9.4.

The Air Conditioning (A/C), Heating, Cooling, and Ventilation Systems are designed to accomplish the following performance objectives:

a) Remove the normal heat gain from the outdoors, equipment, lighting, and people
b) Replace the normal heat lost to the outdoors
c) Provide adequate ventilation for access requirements, and
d) Reduce the concentration of airborne radionuclides, non-radioactive particulate matter, and noxious gases.

The overall system is divided into various component systems as described in Section 9.4. The basic design of flow paths and equipment arrangements is predicated on the criterion of controlling the direction of airflow to ensure that potentially contaminated areas are maintained at negative pressures and the exhaust therefrom is directed to the plant vent for discharge. Section 9.4 describes each component system in detail.

CHAPTER 4
REACTOR

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4.0 REACTOR

4.1 SUMMARY DESCRIPTION

4.1.1 GENERAL DESCRIPTION OF CORE

The H. B. Robinson Unit 2 (HBR 2) reactor core is comprised of an array of 157 fuel assemblies. The core is cooled and moderated by light water at a normal operating pressure of 2250 psia in the Reactor Coolant System (RCS). The Reactor Coolant System contains boron as a neutron poison. The concentration of boron in the reactor coolant is varied as required to control relatively slow reactivity changes including the effects of fuel burnup.

The reactor core and reactor vessel internals are shown in elevation in Figures 4.1.1-1, 4.1.1-2 and a typical loading pattern is shown in Figure 4.1.1-3.

The HBR 2 reactor core contains 157 fuel assemblies manufactured by Advanced Nuclear Fuel Corporation (ANF). Each assembly normally contains 204 fuel rods, twenty rod cluster control (RCC) guide tubes, and one instrumentation tube in a 15 x 15 fuel rod array. The standard fuel rods consist of slightly enriched UO_2 pellets inserted into Zircaloy tubes. Beginning with Region 11 fuel, integrated burnable absorber rods have been used in varying numbers in the core, in the form of rods containing trace amounts of gadolinia (Gd_2O_3) in various concentrations in UO_2, for peaking control and reduction of the BOC critical boron. The Region 14 fuel contains 40 gad-bearing assemblies using 4% gadolinia and 8 assemblies using 6% gadolinia, while the Region 15 fuel includes 32 pins at 10% gadolinia in a total of 8 assemblies. The RCC guide tubes and the instrumentation tubes are also made of Zircaloy. Each assembly contains seven spacers; six of which are located within the active fuel region. In the older design, all of these are bi-metallic. Starting with Cycle 14 fuel, only the bottom spacer is bi-metallic, the rest are a High Thermal Performance (HTP) zircaloy design. There are also three Intermediate Flow Mixer (IFM) grids and a debris-resistant lower tie plate on the HTP fuel.

There are two features that reduce the fast neutron fluence reaching the pressure vessel wall: axially blanketed fuel, and Part Length Shield Assemblies (PLSAs). The axial blanketed fuel contains a region of natural uranium at the top and at the bottom of each fuel assembly. All gadolinia-bearing pins contain 12 inches of natural uranium at the top and bottom of the fuel pin, while the non-gad pins contain 6 inches of natural uranium at the top and bottom. The PLSAs contain a steel insert in the bottom of each fuel rod and a natural uranium blanket for the top six inches of the active core. The steel insert at the bottom reduces the active fuel length, but has no effect on the outside dimensions of the assembly.

TABLE 4.1.2-1

MECHANICAL DESIGN VALUES

A. FUEL PELLETS

Initial Enrichment, wt% U-235	0.711 to 4.20
Form	right cylinder
Average UO_2 Density, % Theoretical	94
Pellet Diameter, in.	0.3565

B. FUEL ROD

Number of Rods per Assembly	204
Active Length, in.	144.0 (102.0 for PLSA rod)
Overall Rod Length, in.	152.065
Rod Pitch, in.	.563
Fill Gas	Helium

C. CLADDING

Material	Zircaloy-4
Outside Diameter, in.	.424
Wall Thickness, in.	.030

D. FUEL ASSEMBLY

Geometry	15 x 15
Number of Assemblies	157 (12 PLSAs, 145 non-PLSAs)
Fuel Assembly Pitch, in.	8.466
Overall Length, in.	159.71 (excluding upper tie plate leaf spring)

E. CONTROL ROD GUIDE TUBE

Number/Assembly	20
Material	Zircaloy-4
ID, Upper Section, in.	.511
ID, Dashpot, in.	.455
Dashpot Length, in.	24.5

F. INSTRUMENTATION TUBE

Number/Assembly	1
Material	Zircaloy-4
ID, in.	.511

TABLE 4.1.2-1 (Cont'd)

MECHANICAL DESIGN VALUES

G.	BI-METALLIC SPACER GRIDS	
	Number: HTP/non-HTP	1/7
	Material	Zircaloy-4/Inconel 718
H.	HTP SPACER GRIDS	
	Number per HTP assembly	6
	Material	Zircaloy-4
I.	IFM Grids	
	Number per HTP assembly	3
	Material	Zircaloy-4
J.	COMPONENT WEIGHTS	
	Weights per Assembly:	
	Fuel	1080 lb (Non-PLSAs, Non-Gad)
	Cladding and End Caps	255 lb
	Bi-metallic Spacers	
	Zircaloy-4	1.42 lb/spacer
	Inconel	0.20 lb/spacer
	Control Rod Guides	21 lb
	Other Hardware	40 lb
	Total Assembly Weight, lb	1417 (Non-PLSAs)
	Uranium Weight per Rod, kg	2.1177 (Non-gad)
	Uranium Weight per Rod, kg	2.0236 (4 w/o gad rod)
	Uranium Weight per Rod, kg	1.9774 (6 w/o gad rod)
	Uranium Weight per Rod, kg	1.8871 (10 w/o gad rod)
	Uranium Weight per Assembly, kg	432.0 (Non-PLSAs, non-gad)
		431.8 (2 gad rods @ 4 w/o)
		431.6 (4 gad rods @ 4 w/o)
		430.9 (12 gad rods @ 4 w/o)
		430.7 (8 gad rods @ 4 w/o - 4 gad rods @ 6 w/o)
		430.0 (8 gad rods @ 6 w/o - 4 gad rods @ 10 w/o)
K.	INSERT USED WITH PLSA FUEL RODS (ONLY)	
	Material	304 stainless steel
	Diameter, In.	0.350
	Length, In.	42.0

Note: The values in Sections J and K are from Cycle 12 and are representative.

TABLE 4.1.2-2

THERMAL-HYDRAULIC DESIGN VALUES 3

Rated Heat Output, Mwt	2300
Maximum Overpower, %	12
Heat Generated in Fuel, %	97.4
Nominal Design Pressure, psia	2250
Nominal Inlet Temperature, °F	546.5
Average Core Temperature, °F	578.0
Nominal Outlet Temperature at Hot Channel, °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	14.3
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*	$19.7 \pm .8$

3

3

3

*Includes static head.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							PLSA N47	PLSA N51	PLSA N47						
2					14 P01	** P33	* P41	H11 N19	* P42	** P34	14 P02				
3				H8 M48	*** P20	K5 N09	J3 M01	**** P30	G3 M02	F5 N10	*** P21	H7 M17			
4			J3 M26	*** P14	F14 M20	E13 N43	G14 N23	G6 M35	J14 N34	L13 N44	K14 M21	*** P15	H4 M49		
5		14 P05	*** P10	B10 M24	H4 N37	E12 L36	E14 N03	J10 M34	L14 N04	L12 L35	C4 N38	P10 M25	*** P11	14 P06	
6		** P37	L6 N13	C11 N35	D11 M11	**** P26	G12 M43	*** P17	J12 M44	**** P25	M11 M12	N11 N36	E6 N14	** P38	
7	PLSA N54	* P45	N7 M13	B9 N30	B11 N07	D9 M52	J7 M32	H14 N27	G7 M31	M9 M45††	P11 N08	P9 N24	C7 M14	* P46	PLSA N48
8	PLSA N50	E8 N18	**** P28	K9 M39	F7 M38	*** P16	B8 N26	H8 M53	P8 N28	*** P18	K7 M40	F9 M37	**** P29	L8 N20	PLSA N52
9	PLSA N46	* P48	N9 M16	B7 N22	B5 N06	D7 M51	J9 M30	H2 N25	G9 M29	M7 M46	P5 N05	P7 N32	C9 M15	* P47	PLSA N56
10		** P40	L10 N16	C5 N34	D5 M10	**** P31	G4 M42	*** P19	J4 M41	**** P27	M5 M09	N5 N33	E10 N15	** P39	
11		14 P08	*** P9	B6 M23	M12 N40	E4 L12	E2 N02	J6 M36	L2 N01	L4 L42	D12 N39	P6 M22	*** P24	14 P07	
12			H12 M47	*** P13	F2 M19	E3 N42	G2 N29	G10 M33	J2 N21	L3 N41	K2 M18	*** P12	G8 M27		
13				H9 M28	*** P23	K11 N12	J13 M04	**** P32	G13 M03	F11 N11	*** P22	D8 H50			
14					14 P04	** P36	* P44	H5 N17	* P43	** P35	14 P03				
15							PLSA N53	PLSA N49	PLSA N45						

* 4 Pins of 4 w/o Gadolinia Per Assembly

** 8 Pins of 4 w/o Gadolinia per Assembly in Region 14 Fuel

*** 12 Pins of 4 w/o Gadolinia Per Assembly in Region 14 Fuel

PLSA Part Length Shield Assembly

**** 12 Pins of 6 w/o Gadolinia Per Assembly in Region 14 Fuel

†† This assembly contains one (1) inert zirconium rod

Fresh Gadolinia-Bearing Assembly
Cycle 10 core location or region number
Assembly fabrication ID

AMENDMENT NO. 6

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

TYPICAL LOADING PATTERN:
CYCLE 11 FOR AN EOC 10 EXPOSURE OF
11,150 MWD/MT

FIGURE
4.1.1-3

4.2 FUEL SYSTEM DESIGN

4.2.1 DESIGN BASES

4.2.1.1 Summary

The fuel design for the H. B. Robinson plant has been modified starting with the ANF-11 Reload (Cycle 14, starting in 1991). The new fuel is a High Thermal Performance (HTP) design. This fuel incorporates six all-zircaloy HTP spacers, three Intermediate Flow Mixers (IFMs), one bi-metallic bottom spacer, and a debris-resistant lower tie plate. The design assembly burnup has been increased to 52.5 GWd/Mtu. The fuel rod design is unchanged, although in the rods which contain gadolinia poison, the natural UO_2 blankets are 12, rather than 6, inches in height. There are 4, 6, or 12 gadolinia rods per assembly in those ANF-11 assemblies containing the neutron poison.

Mechanical design analyses were performed to evaluate cladding steady-state strain, transient stress and strain, fatigue, creep collapse, corrosion, hydrogen absorption, fuel rod internal pressure, elongation, and fuel assembly growth. Design criteria consistent with current ANF methodology were used in the analysis. For the non-HTP fuel, a peak rod burnup of 50,000 MWD/MTU for the axial blanket fuel rods, and a peak assembly burnup of 44,000 MWD/MTU, were analyzed. Bounding power histories have been used. The results indicate that all the mechanical design criteria are satisfied.

a) The maximum end-of-life (EOL) steady-state cladding strain was determined to be negative, thus meeting the 1.0 percent design limit.

b) The cladding stress and strain during power ramps, calculated under different overpower conditions, do not exceed the design stress corrosion cracking threshold or the 1.0 percent strain limit.

c) The cladding fatigue usage factor is within the design limit.

d) The end-of-life fuel rod internal pressure is less than the approved design limit.

e) The criterion for the prevention of creep collapse is satisfied.

f) The maximum calculated EOL thickness of the oxide corrosion layer and the maximum calculated concentration of hydrogen in the cladding are within the design limits.

4.2.1.2 Fuel Rod Design Basis

4.2.1.2.1 Cladding Physical and Mechanical Properties

Zircaloy-4 combines a low neutron absorption cross section, high corrosion resistance, and high strength and ductility at operating temperatures. Principal physical and mechanical properties including irradiation effects on Zircaloy-4 are provided in Section 4.2.2.3.

4.2.1.2.2 Cladding Stress Limits

The design basis for the fuel cladding stress limits is that the fuel system will not be damaged due to fuel cladding stresses exceeding material capability. Conservative limits shown in Table 4.2.1-1 are derived from the ASME Boiler and Pressure Vessel Code, Section III, Article III-2000 (Reference 4.2.1-1).

The cladding may also be damaged by the combination of volatile fission products and high cladding tensile stresses which may lead to stress corrosion cracking. Stress corrosion cracking of fuel rod cladding is considered the principal failure mechanism for pellet-cladding interaction (PCI) failures encountered during changes in reactor operating conditions.

The concept used to avoid failures from the stress corrosion crack failure mechanism from power ramps is to keep the fuel rods from operating above the stress threshold associated with the nucleation of a propagating stress corrosion crack.

4.2.1.2.3 Cladding Strain Limits

Tests on irradiated tubing (References 4.2.1-2 and 4.2.1-3) indicate potential for failure at relatively low mean strains. The data on tensile, burst and split ring tests, indicate a ductility ranging between 1.2 percent and 5 percent at normal reactor operating temperatures. The failures are usually associated with unstable or localized regions of high deformation after some uniform deformation. To prevent cladding failure due to plastic instability and localization of strain, the total mean hoop cladding strain for steady-state conditions is limited to 1 percent, and the increment of the thermal creep during a transient is also limited to 1 percent.

4.2.1.2.4 Strain Fatigue

Cyclic PCI loading combined with other cyclic loading associated with relatively large changes in power can cause cumulative damage which may eventually lead to fatigue failure. Cyclic loading limits are established to prevent fuel failures due to this mechanism. The design life is based on correlations which give a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles whichever is more conservative (Reference 4.2.1-4).

4.2.1.2.5 Fretting Corrosion and Wear

The design basis for fretting corrosion and wear is that fuel rod failures due to fretting shall not occur. Since significant amounts of fretting wear can eventually lead to fuel rod failure, the spacer grid assemblies are designed to prevent such wear.

4.2.1.2.6 Corrosion

Cladding oxidation and corrosion product buildup are limited in order to prevent significant degradation of clad strength. A PWR clad external temperature limit is chosen so that corrosion rates are very slow below this temperature and therefore overall corrosion is limited. An external corrosion layer limit is also specified so that this amount of corrosion will no

4.2.2 DESIGN DESCRIPTION

4.2.2.1 Fuel Assembly

The 15x15 fuel assembly array includes 20 guide tubes, 204 fuel rods and one instrumentation tube. Six of the seven grid spacers are an all-zircaloy High Thermal Performance (HTP) design. The bottom spacer grid is bi-metallic. There are three Intermediate Flow Mixer (IFM) grids, which along with the HTP grids, have internal slanted channels that improve the fuel rod heat transfer and coolant mixing. The fuel assembly tie plates are stainless steel castings with Inconel holddown springs. Fuel assembly characteristics are summarized in Table 4.2.2-1. The existing and HTP fuel assemblies are shown in Figures 4.2.2-1 and 4.1.2-3.

The grid spacers are welded to the Zircaloy-4 guide tubes, and the guide tubes are mechanically attached and secured to the upper and lower tie plates. The instrumentation tube is mechanically captured between the tie plates and welded to the grid spacers. The fuel rods are axially positioned within the skeleton with approximately equal spacing at both ends. The upper tie plates are designed to be removed and reinstalled by underwater remote handling techniques.

Proper orientation of fuel assemblies is specifically addressed through the design of the upper tie plate. As shown in Figure 4.2.2-1, it has two locating holes in opposite corners for receiving the locating pins in the upper core support plate. A third hole of smaller diameter is located in a third corner for the purpose of orienting the assembly. This hole receives the indexing pin from the manipulator grapple.

4.2.2.1.1 Fuel Assembly Material Properties

The material properties used in the design evaluation are described in this section.

4.2.2.1.2 Zircaloy-4 Chemical Properties

Zircloy-4 is used in three forms: (a) cold worked and stress relieved cladding; (b) recrystallized annealed tubing; and (c) recrystallized annealed strip.

4.2.2.1.3 Fissile Material (Uranium Dioxide)

Chemical composition is as follows:

a) Uranium Content - The uranium content shall be a minimum of 87.7 percent by weight of the uranium dioxide on a dry weight basis.

b) Stoichiometry - The oxygen-to-uranium ratio of the sintered fuel pellets shall be within the limits of 1.99 and 2.01.

Mechanical properties are as follows:

a) Mechanistic Fuel Swelling Model - The irradiation environment and fissioning events cause the fuel material to alter its volume and, consequently, its dimensions.

b) Fission Gas Release - For design evaluations of end-of-life pressures, pellet-cladding interaction and general thermal mechanical conditions, a

physically based two-stage release model is used. First stage fission gas release is to grain boundaries, and then the second stage release is from the grain boundaries to the interconnected free gas volume.

c) Melting Point - The value used for the UO_2 melting point (unirradiated) is 2805°C (5081°F). Based on measurements by Christensen, et al (Reference 4.2.2-1), the melting point is reduced linearly with irradiation at the rate of 12.2°C (22.0°F) per 10^{22} fiss/cm^3 or 32°C (57.6°F) per 10^4 MWd/MTU.

4.2.2.1.4 Inconel Springs

Coil springs are fabricated from Inconel X-750 wire or rod with an alloy composition in accordance with AMS 5699B. Leaf springs are fabricated from Inconel sheet or strip.

4.2.2.2 Fuel Rod

The fuel rods consist of cylindrical UO_2 pellets in Zircaloy-4 tubular cladding.

The Zircaloy-4 fuel rod cladding is cold worked and lightly stress relieved. Zircaloy-4 plug type end caps are seal welded to each end. The upper end cap has external features to allow remote underwater fuel rod handling. The lower end cap has a truncated cone exterior to aid fuel rod reinsertion into the fuel assembly during inspection and/or reconstitution.

Each non-PLSA fuel rod contains a 132.0 inch column of enriched UO_2 fuel pellets, and a 6 inch column of natural UO_2 fuel pellets at each end except for gadolinia bearing fuel rods which have a 12-inch natural uranium blanket at the top and bottom of the fuel rod. Each PLSA fuel rod has the bottom 42 inches of fuel replaced by stainless steel.

The fuel rod upper plenum contains an Inconel compression spring to prevent fuel column separation during fabrication and shipping, and during in-core operation.

Fuel rods are pressurized with helium which provides a good heat transfer medium and assists in the prevention of clad creep collapse. The fuel rod is shown in Figure 4.2.2-2.

4.2.2.3 Core Components

4.2.2.3.1 Rod Cluster Control Assembly

The RCCA are provided to control the reactivity of the core under operating conditions. These assemblies, one of which is shown in Figure 4.2.2-3, each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. RCCA details are presented in Table 4.2.2-2.

The absorber material used in the control rods is silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCCA are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive shaft are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the RCCA and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub and the fingers are joined to the vanes by furnace brazing. A centerpost which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from Type 304 Stainless Steel, except for the springs, which are Inconel X-750 alloy, and the retainer, which is of 17-4 PH material.

The absorber rods are secured to the spider so as to assure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver-indium-cadmium rods are inserted into coldworked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral and end clearances are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions.

4.2.2.3.2 Neutron Source Assembly

The H. B. Robinson core normally utilizes one to two neutron source assemblies. Historically, these sources have been composed of four secondary source rods, however, beginning in Cycle 14 source assemblies with eight secondary source rods will be used to increase source strength (this does not preclude a return to sources with four secondary rods in the future). The increased source strength is necessary to overcome the shielding effect of the PLSA assemblies which are located between the sources and the source range detectors. The rods in the secondary source assemblies (both 4 and 8 finger) are fastened to a spider-hub at the top similar to a rod cluster control assembly (RCCA) spiders. In the core, the neutron sources assemblies are inserted into fuel assembly guide tubes at locations that are unrodded and with which there will be mechanical compatibility between the spider-hub and the reactor upper internals. Figure 4.3.2-1 illustrates the preferred secondary source locations of H-03 and H-13.

General design criteria similar to that for the fuel rods are used for the design of the source rods; i.e., the cladding is free-standing, internal

pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and cladding. Typically, secondary source rods used at H. B. Robinson have utilized cold-worked Type 304 Stainless Steel cladding material (nominal 0.431 in. OD, 0.3935 in. ID) with Sb-Be source pellets of stack height 67.87 in. Alternative designs are possible provided they meet the general design criteria.

In some cases more than two source assemblies may be used in the core to provide an active source during startup while transitioning from old previously irradiated sources to new inactive sources; at the completion of a "source transition cycle" the old sources are typically removed and disposed of. In this circumstance, some source assemblies must be located in core locations other than the preferred locations H-03 and H-13. The following alternative core locations provide mechanical compatibility between the reactor upper internals and the spider-hub type source assemblies utilized at H. B. Robinson:

A-07	A-09	B-07	B-09	B-11	C-04	C-05	C-06
C-10	C-11	D-07	D-09	D-11	D-13	E-02	E-03
E-06	E-13	F-07	G-04	G-10	G-12	G-14	H-01
H-03	H-07	H-13	J-01	J-04	J-06	J-08	J-14
J-15	K-07	K-09	K-13	L-02	L-03	L-04	L-10
L-13	M-05	M-07	M-13	N-05	N-11	N-12	P-05
P-09	P-11	R-08	R-09				

If the purpose of a given source located in a core position other than H-03 or H-13 is to provide counts for the source range detectors, acceptable (but not necessarily exclusive) alternative locations taken from the mechanically compatible list are:

G-04 G-12 G-14 J-04 J-14

In locating a new inactive source for irradiation and use in the following cycle, an additional consideration in choosing its location is that the host assembly should experience a relative power of at least 0.5 to provide sufficient activation.

4.2.2.3.3 Thimble plug assembly. In order to limit bypass flow through the RCC guide thimbles in fuel assemblies which do not contain either control rods or source assemblies, the fuel assemblies at those locations are fitted with plugging devices. The plugging devices consist of a flat spider plate with short rods suspended from the bottom surface and a spring pack assembly attached to the top surface. At installation in the core, the plugging devices fit with the fuel assembly top nozzles and rest on the adaptor plate. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core when the upper internals package is lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles.

All components in the plugging device, except for the springs, are constructed from Type 304 Stainless Steel. The springs are wound from an age hardenable nickel-base alloy.

TABLE 4.2.2-1

FUEL ASSEMBLY DESIGN

FUEL PELLET	
Fuel Material	UO_2 Sintered Pellets
Pellet Diameter, (in.)	0.3565
CLADDING	
Clad Material	Zircaloy-4 Cold Worked and Stress Relieved
Clad ID, (in.)	0.364
Clad OD, (in.)	0.424
Clad Thickness, Nominal, (in.)	0.030
FUEL ROD	
Diameter Gap, Cold Nominal, (in.)	0.0075
Active Length, (in.)	144.0
Total Rod Length, (in.)	152.065
Fill Gas	Helium
BI-METALLIC SPACER	
Material	Zr-4 & Inconel 718
Envelope (in.)	8.426 square
HIGH THERMAL PERFORMANCE (HTP) SPACER	
Material	Zircaloy-4
Envelope (in.)	8.426 square
INTERMEDIATE FLOW MIXER (IFM) GRID	
Material	Zircaloy-4
Envelope (in.)	8.395 square

4.2.3 MECHANICAL DESIGN EVALUATION

4.2.3.1 Reactor Operating Conditions for Design

The fuel assembly design is based on the following reactor operating conditions:

Core Power Level	
Nominal	2300 MWT
Design Basis (12% Thermal Overpower)	2576 MWT
Coolant Operating Pressure (Nominal)	2250 Psia
Coolant Flow Rate (Min. @ Nominal Power)	
Total	101.5 X 10^6 lb/hr
Active Core	97.0 X 10^6 lb/hr
Heat Generation Fraction Fuel Rods	97.4 percent
Average Flow Velocity	14.3 ft/sec
Coolant Inlet Temperature (Nominal)	546.5°F
Core Average Coolant Temperature	578°F
Number of Assemblies in Core	157

The fuel shall be capable of load-follow operation between 50 percent and 100 percent of rated power, and not preclude the transients set forth in the UFSAR.

4.2.3.2 Fuel Rod Evaluation

4.2.3.2.1 Design Criteria

a) Cladding steady state stresses shall not exceed the established limits.

b) Maximum cladding strain shall not exceed 1.0 percent at end-of-life (EOL), or 1.0 percent during power transients.

c) During power transients, the maximum hoop stress in the cladding shall be limited to avoid failure by stress corrosion cracking.

d) The cumulative usage factor for cyclic stresses shall not exceed 0.67.

e) The fuel rod internal pressure at the end of the design life may exceed the system operating pressure up to the NRC approved design limit.

f) Cladding creep collapse shall not occur.

g) The hydrogen absorption of the cladding and the thickness of the corrosion layer shall not exceed design limits.

h) The fuel elongation must be accommodated by the clearance between fuel rods and tie plates.

i) Fuel rod creep bow throughout the design life of the assemblies shall be limited so as to maintain licensing and operational limit restraints.

j) The fuel rod plenum spring shall maintain a positive compression on the fuel column during shipping and during the fuel densification stage.

k) Cladding temperatures shall not exceed the design limits.

l) Pellet temperatures shall not exceed the melting temperature during normal operation and anticipated transients.

4.2.3.2.2 Fuel Rod Analysis

The fuel rod analysis considers the high burnup design with natural uranium axial blankets. The analysis does not specifically consider the gadolinia bearing fuel rods, since the neutronic criteria require a reduced enrichment, such that the fuel temperatures of the neutron absorbing fuel (NAF) pellets shall be less than the UO_2 pellets during limiting fuel cycle operation. The analyses described in this Section are detailed and documented in References 4.2.3-1 and 4.2.3-2.

a) Steady State Stresses - The cladding steady-state stresses are highest at beginning-of-life except for a bending stress due to ovality. Since the cladding eventually is supported by the pellets, the ovality bending stress is eliminated as a factor for the end-of-life condition at higher burnup. The cladding stresses are within the established limits.

The stress analysis is performed at the lower end cap since the maximum temperature gradients occur at this end.

The mechanical stress is caused by the pressure differential across the rod wall and by the axial load of the pellet stack weight and the plenum spring force. The thermal stress is caused by the temperature gradient between the end cap and the heat generating pellets.

The ANSYS code, which allows thermal as well as stress analyses, was used to model the subject rod region. The maximum weld stress intensity is well below the design limit.

b) Steady State Strain Analyses - The cladding steady-state strain was evaluated with the RODEX2 code. The code calculates the thermal, mechanical and compositional state of the fuel, and cladding for a given duty history. Conservative input values were used in the strain analysis. Bounding dimension values covering all reloads were selected for the calculations.

The criterion of 1 percent maximum at EOL is satisfied.

c) Ramp Stress and Strain Analysis - The clad response ramping power changes is calculated with the RAMPEX code. This code calculates the pellet-cladding interaction during a power ramp. The initial conditions are obtained from RODEX2 output. The RAMPEX code considers the thermal condition of the rod in its flow channel and the mechanical interactions that result from fuel creep, crack healing, and cladding creep at any desired axial section in the rod during the power ramp.

The power histories assumed for this analysis are the maximum exposure rods and arbitrary power histories with low and intermediate powers for the first cycle, followed by high power second and third cycles. The arbitrary cycles were used to evaluate large power swings resulting from fuel shuffling.

The conditions at the end of each cycle obtained with the RODEX2 code are used as input data for the RAMPEX code. The rods under consideration were ramped to the maximum power. The peak stresses obtained are below the threshold stress for stress corrosion cracking, thus complying with the design requirements. In addition, the maximum clad strains due to each ramp were examined. The maximum strains, including primary and secondary thermal creep, were less than the 1 percent strain limit.

d) Cladding Fatigue Usage Factor - In addition to the ramp strain analyses, a fatigue usage factor for the cladding was calculated. The calculations were based upon the typical duty cycles. Cladding stress amplitudes for the various power cycles were determined from RAMPEX analyses. RAMPEX analyses were run for each cycle at the plane of maximum contact pressure which resulted in conservatively high stresses for the fatigue analysis. The overall fatigue usage factor is within the 0.67 design limit.

e) Internal Pressure - A RODEX2 analysis was performed to evaluate the end-of-life (EOL) internal fuel rod pressure. To prevent cladding instability, the rod internal pressure cannot exceed the approved design limit or else the cladding may creep away from the pellet, which increases the fuel rod pellet temperatures. Higher fuel temperatures result in increased fission gas release and, therefore, higher internal rod pressures. The results of this analysis show the EOL internal rod pressure does not exceed the system pressure of 2250 psia. The fuel rod will, therefore, remain stable throughout the expected power history.

f) Creep Collapse - The collapse calculation is done using the RODEX2 and COLAPX codes to determine the temperature and pressure conditions throughout the fuel rod lifetime, and to determine the clad creepdown. These conditions are used as input for COLAPX. The COLAPX code then predicts the time dependent creep ovality deformations in an infinite length tube subjected to external pressure, internal pressure, and linearly varying temperature gradients through the thickness of the cylinder.

If significant gaps are not allowed to form, then tube ovality, as predicted by the COLAPX evaluation, cannot occur beyond the point of fuel support.

In order to guard against the highly unlikely event that enough densification occurs to form pellet column gaps of signficant size to allow clad flattening, an evaluation was performed. The cladding ovality increase was calculated with COLAPX, and the creepdown was calculated with RODEX2. The combined creepdown at the cladding minor axis was determined not to exceed the minimum level to allow the fuel column to relocate axially without the formation of axial gaps.

g) Rod Bowing - Fuel rod bow is determined throughout the life of the fuel assembly so that reactor operating thermal limits can be established. These limits include the minimum critical heat flux ratio associated with protection against boiling transition and the maximum fuel rod LHGR associated with protection of metal-water reaction and peak cladding temperature limits for a postulated loss of coolant accident (LOCA).

Rod bow measurements have been used to establish an empricial model for determining rod bow as a function of burnup which is used to calculate thermal limits.

The gap spacing data shows that the bow tends to stabilize at higher burnups. In addition, the fuel at high burnups is not limiting from a thermal margin standpoint due to its lower power.

h) Corrosion Layer and Hydrogen Absorption Analyses - The thickness of the corrosion layer and the amount of hydrogen absorbed by the cladding have been evaluated with the RODEX2 code for the peak discharge fuel rod power history. The oxide thickness and the hydrogen content are well below the design limits.

i) Fuel Rod Growth - Growth data has been correlated to fast fluence. Based on this correlation, with an added uncertainty, the rod growth for the maximum discharge exposure fuel rod was calculated. Conservatively, assuming no tie plate spacing guide tube growth, a minimum end of life clearance margin for this growth is available. An additional 0.2 inch allowance for fuel rod growth was created in the HTP assemblies by taking that much off the lower tie plate legs and using it as space between the tie plates. The overall guide tube length was also increased by 0.2 inch.

j) Cladding Temperature - Prevention of potential fuel failure from overheating of the cladding is also established by minimizing the probability that DNB occurs on limiting fuel rods during normal operation and anticipated operating events.

k) Fuel Pellet Temperature - Prevention of fuel failure from overheating of the fuel pellets is accomplished by insuring that the peak LHGR during normal operation and anticipated transients does not result in calculated centerline melt.

4.2.3.3 Fuel Assembly Evaluation

4.2.3.3.1 Design Criteria

The mechanical design criteria for the fuel assembly are listed below:

a) The fuel assemblies shall be mechanically compatible with the reactor core, fuel handling system, and core components.

b) The upper tie plate shall be removable from the fuel assembly to permit access for removal of fuel rods for replacement or inspection.

c) The fuel assembly shall be designed to withstand operating, handling, and accident loads.

d) The fuel assembly shall support the fuel rod, providing sufficient spring force to minimize flow-induced vibrations and to prevent fretting corrosion at the spacer-fuel rod contact points.

e) The assembly shall be designed to provide clearance for irradiation induced guide tube growth without exceeding the core plate-to-core plate spacing.

4.2.3.3.2 Fuel Assembly Analysis

a) Stresses and Deflections

The guide tubes along with the upper and lower tie plates and grid spacers provide the principal structure for the fuel assembly. Guide tubes are considered as restrained columns and are analyzed accordingly, using appropriate load combinations. Column deflection is permissible within constraints of allowable bending stress, allowable displacement, and allowable approach to column instability. The allowable total stress, primary plus bending, is less than the yield strength of the material at the temperature of the load conditions (Reference 4.2.3-3).

As the power level of the reactor is increased, differential thermal expansion between the Zircaloy guide tubes and the hotter Zircaloy clad fuel rod would tend to put the guide tube in tension. Therefore, there is no concern as to the stability of the guide tube on approach to normal operating conditions. After some period at power, vibration loads would tend to reduce or eliminate loads caused by differential thermal expansion. Upon reduction in power, differences in temperature between the guide tubes and fuel rods would decrease causing compression loading on the guide tube. Thus, the stability of Zircaloy guide tubes is of most concern as the power level is reduced.

The Zircaloy spacer was analyzed using a finite element structures code. The structural integrity was confirmed through strength tests. Some tests used a hydrided spacer in order to simulate in-reactor conditions.

The most severe normal loading condition is the situation where the lower tie plate becomes hung up on a spacer edge during fuel handling. Both analyses and tests indicate that the spacer structure can take such loading.

Cyclic loading due to differential thermal expansion between the fuel rods and guide tubes is less severe than the assumed refueling load described above. In this latter case the maximum load is uniformly distributed across the spacer structure as compared to the refueling situation load which is concentrated at local regions at the spacer edge. Thus, loading due to differential thermal expansion of the structure should not result in stresses sufficient to cause fatigue failures.

b) Fuel Rod Support

The Inconel spacer springs are known to relax during irradiation and the fuel rod cladding tends to creepdown. Together, these two characteristics combine to reduce the spacer spring force on a fuel rod during its lifetime. These characteristics have been considered in the design of the spring to assure an

adequate holding force when the assembly has completed its design operating life.

The prevention of fretting corrosion in the HTP and IFM spacers is demonstrated by a combination of analysis and fretting tests. The design analysis determines the projected maximum end-of-life gap, considering spring relaxation, clad creepdown, minimum fuel rod outer diameter, and minimum initial spring deflection. Flow test data are used to confirm that fretting corrosion will not occur for the largest possible projected gaps.

c) Fuel Assembly Growth

The limiting condition for fuel assembly growth is at end-of-life after cooldown. Because of the higher coefficient of thermal expansion for the stainless steel core structure relative to the Zr-4 guide tubes, differential thermal expansion increases the assembly/internals structure clearance during heatup and reduces the clearance upon cooldown. The guide tube growth data for ANF irradiated fuel assemblies includes data for the H. B. Robinson extended burnup demonstration to 47.7 GWD/MTU. Allowing for measurement error and other uncertainties, the maximum EOL fuel assembly length predicted from the upper limits of the data leaves a clearance with the minimum as-built core plate to core plate separation.

d) Combined Shock and Seismic Loading (Internals)

The results of a detailed study of the blowdown plus seismic excitation of the reactor internal indicated that the maximum deflections and stresses in the critical structures are below the established allowable limits. For the transverse excitation, it was shown that the upper barrel would not buckle during a hot-leg break and that it would have an allowable stress distribution during a cold-leg break. Even though control rod insertion is not required for plant shutdown, the analysis shows that none of the guide tubes will deform beyond the "no loss of function" limits established experimentally for control rod insertion, and 52 out of 53 guide tubes would deform less than the conservatively established allowable limit. Consequently, it is concluded that the reactor internals will be able to withstand the assumed accident conditions without becoming distorted enough to prevent adequate core cooling or reactor shutdown.

e) Combined Shock and Seismic Loading (Fuel Assembly)

The reload fuel was evaluated for combined seismic and loss-of-coolant accident (LOCA) mechanical response (Reference 4.2.3-4). The postulated accident condition considered was a 0.2-g seismic event combined with a 144-square-inch pipe break at the cold leg reactor pressure vessel inlet nozzle.

The lateral core plate motions for the seismic and LOCA events were combined based on maximum fuel assembly loads and displacements. The vertical forces from the pipe break at the cold leg reactor pressure vessel inlet nozzle were determined from a summation of pressure differentials acting across a given element, flow stagnation, orifice losses, and friction losses. In addition to these hydraulic forces, gravity forces, buoyancy forces, and holddown spring preload were also included in the analysis.

4.2.4 TESTING AND INSPECTION PLAN

4.2.4.1 Quality Assurance Program

Information on the CP&L Quality Assurance Program is provided in Chapter 17 of the updated FSAR. The Exxon Nuclear Quality Assurance Program is described in XN-NF-1 (Reference 4.2.4-1).

4.2.4.2 Quality Control

Fuel assembly quality control is achieved by a component inspection program which has the following features:

a) An enrichment verification program which covers incoming UF_6 gas to completed fuel rods

b) Verification of cladding integrity by testing and inspection of each lot of tubing received

c) Inspection of fuel pellets for conformity to specification

d) Radiographic examinations

e) Inspection of each fuel assembly for cleanliness, straightness, envelope, rod-to-rod spacing, length and fuel rod axial position, and

f) On-site inspection for fuel rod axial position, rod-to-rod spacing and cleanliness.

Details of the fuel assembly quality control activities are provided in XN-75-39 (Reference 4.2.3-3).

4.2.4.3 Incore Control Component Testing and Inspection

To confirm the mechanical adequacy of the fuel assembly and RCCA, functional test programs have been conducted on a full scale San Onofre mock-up version of the fuel assembly and control rods. Additional tests were run on two full scale prototype assemblies for a 12 ft active core. One of the 12 ft assemblies incorporated stainless steel guide tubes and the other incorporated Zircaloy-4
tubes.

4.2.4.3.1 Reactor Evaluation Center Tests

The prototype assemblies were tested under simulated reactor operating conditions (1875 psig, 575°F, and 17.8 fps flow velocity) in the Westinghouse Reactor Evaluation Center for a total of more than 6400 hours.

Each prototype assembly was subjected to scram cycling equivalent to one or more plant lifetimes. The test history for each prototype is summarized in Table 4.2.4-1.

Each of the three prototype fuel assemblies remained in excellent mechanical condition. No measurable signs of wear on the fuel tubes or control rod guide tubes were found.

The control rod was also found to be excellent in condition, having maximum wear measured on absorber cladding of approximately 0.001 in.

4.2.4.3.2 Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location were also successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

4.2.4.3.3 Axial and Lateral Bending Tests

In addition, axial and lateral bending tests were performed in order to simulate mechanical loading of the assembly during refueling operation.

Although the maximum column load expected to be experienced in service is approximately 1000 lb, the fuel assembly can successfully be loaded to 2200 lb axially with no damage resulting. This information is also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

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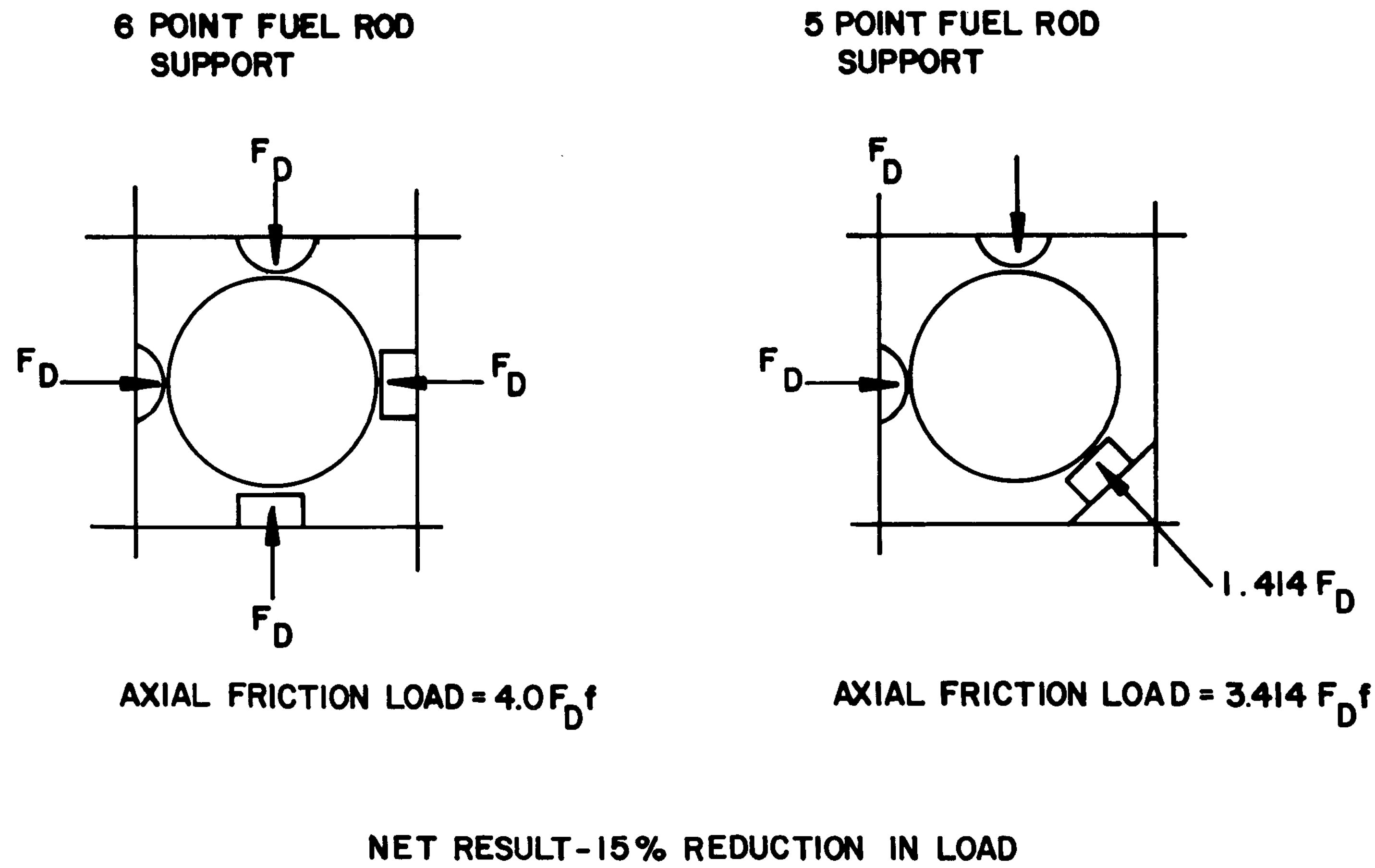
4.2.4-1 XN-NF-1, "Quality Assurance Manual-Nuclear Fuel," October 14, 1981.



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

EFFECT OF SPACER CELL GEOMETRY ON AXIAL RESTRAINT

FIGURE 4.2.3 - 1

4.3 Nuclear Design

4.3.1 Design Basis

Nuclear design bases have been established to assure that the reactor core is operated within the HBR 2 Technical Specifications, Appendix A to Facility Operating License No. DPR-23.

4.3.1.1 Fuel Burnup. The length of each reload cycle shall be determined on the basis of a cycle length consistent with the previous reload burnup. Fuel burnup is restricted by limits on peak assembly burnup, specifically, 44,000 MWD/MTU for the non-HTP fuel and 52,500 MWD/MTU for the HTP fuel.

4.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficients). The initial core and all reload cores are not allowed to have a positive moderator temperature coefficient when operating at full power.

4.3.1.3 Control of Power Distributions. The full loading pattern shall achieve power distributions such that the peak F_Q shall not exceed 2.32 (including uncertainties) in any single fuel rod throughout the cycle under nominal full power operations.

4.3.1.4 Maximum Controlled Reactivity Insertion Rate. The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited so as to preclude rupture of the reactor coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or a rod ejection accident.

4.3.1.5 Shutdown Margins. The fuel loading pattern shall achieve control rod reactivity worths such that the scram worth of all rods minus the most reactive rod shall exceed the beginning of cycle (BOC) and end of cycle (EOC) shutdown requirements.

4.3.1.6 Stability. The protection system ensures that the nuclear core limits are not exceeded during the course of axial xenon oscillations.

4.3.1.7 Emergency Shutdown Capability. Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

4.3.2 DESCRIPTION

4.3.2.1 Nuclear Design Description

The HBR 2 reactor core consists of 157 assemblies, each having a 15 x 15 fuel rod array. Each assembly normally contains 204 fuel rods, 20 rod cluster control (RCC) guide tubes, and one instrumentation tube. The fuel rods consist of slightly enriched (in U-235) UO_2 or UO_2 - Gd_2O_3 pellets inserted into Zircaloy tubes. The uranium enrichment in the gadolinia pins varies roughly inversely with the gadolinia concentration. The RCC guide tubes and the instrumentation tube are also Zircaloy tubes. Each ANF assembly contains 7 Zircaloy spacers with Inconel 718 springs, 6 of which are located within the active fuel region.

The average enrichment for each ANF reload is consistent with the specified reactor energy requirement for the projected effective full power days for that cycle and subsequent cycles. A loading pattern for each cycle is identified which satisfies the criteria on the peak F_{xy} and the largest calculated axial peaking factor. The enrichment of U-235 in the gadolinia pins is reduced relative to that of the non-gadolinia pins to ensure that the gadolinia pins are never the limiting pins in the assembly, even taking into account the reduced thermal conductivity and melting point in these pins due to the gadolinia. For each specified cycle length the calculated end of cycle critical boron concentration is determined.

The excess reactivity control characteristics are determined for each cycle. These include the differential boron worth at full power conditions as a function of cycle lifetime and control rod worths, including the stuck and ejected rod worths. Control rod shutdown margins and reactivity coefficients are also determined for each fuel cycle.

The effective delayed neutron fractions at BOC and EOC are also calculated for each fuel cycle.

Table 4.1.2-3 presents a summary of some key neutronic characteristics for a typical core loading.

4.3.2.2 Power Distributions

Power distribution control is necessitated by reactor safety considerations. The reactor must be capable of safe operation throughout core life, under both steady state and transient conditions, without exceeding acceptable fuel damage limits. If this performance objective is met, the release of unacceptable amounts of fission products to the reactor coolant is prevented.

To this end, two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kW/ft. Second, the minimum departure from nucleate boiling ratio (DNBR) in the core must not be less than the safety limit in normal operation or in short-term transients.

In addition to the above design basis for fuel performance, the initial steady state conditions for the peak linear power for a loss-of-coolant accident (LOCA) must not exceed the values assumed in the accident evaluation (Chapter 15.0). This limit is required in order for the maximum clad temperature attained during a postulated LOCA to remain below that established by the Emergency Core Cooling System (ECCS) Acceptance Criteria.

To aid in specifying the limits on power distribution the following hot channel factors are defined:

a) F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

b) F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

c) F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

d) $F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes through the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNBR at full power are met, provided the values of F_Q and $F_{\Delta H}$ specified in the HBR 2 Technical Specifications are not exceeded. These values are as follows:

a) $F_Q (Z) \leq (2.32/P) \times K(Z)$ for $P>0.5$

b) $F_Q (Z) < (4.64) \times K(Z)$ for $P\leq 0.5$, and

c) $F_{\Delta H}^N < 1.65 [1 + 0.2 (1-P)]$

Where P is the fraction of licensed power at which the reactor core is operating, K(Z) is the normalized axial dependence factor as a function of core elevation as given in the Technical Specifications, and Z is the core height location of F_Q.

In the specified limit of F_Q^N, there is a 5 percent allowance for uncertainties which means that normal operation of the core within the defined conditions and

procedures is expected to result in a measured F_Q^N 5 percent less than the limit for example, at rated power even on a worst case basis. When a measurement is taken, experimental error must be allowed for and 5 percent is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

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 49 dual element bottom-mounted (one element is a spare thermocouple) thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected locations; and 49 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 43 selected fuel assemblies to provide remote reading of the axial flux distribution. 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 118 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 shown in Figure 4.4.3-2.

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 the safety limit (specified in Section 4.4) based on

steady state nominal operating power levels less than or equal to 100 percent, steady state nominal operating RCS average temperature 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.

4.4.3.4 Core Subcooling Monitor

The purpose of the subcooling monitor is to provide a continuous indication of margin to saturated conditions. The monitor uses inputs from core outlet thermocouples, RCS hot and cold leg resistance temperature detectors and RCS system pressure to drive a micro-processor which calculates saturation temperature and determines the margin to saturation based on the inputs. The individual inputs as well as the margin to saturation can be displayed on the monitor's plasma display panel. The monitor has 2 independent channels, and each channel has its own plasma display panel.

4.4.3.5 Digital Metal Impact Monitoring System

The Digital Metal Impact Monitoring System (DMIMS) uses an array of accelerometers externally mounted to the major components to the reactor system, signal conditioning equipment, recording and alarm equipment, and diagnostic equipment and software. This system collects information that may be used by the operator in the detection, location, and identification of loose parts within the reactor coolant system.

4.4.3.5.1 Design Basis

The system components of the DMIMS within the containment are designed and installed to function following all seismic events that do not require plant shutdown (i.e., up to and including OBE). Recording equipment need not function without maintenance following the specified seismic event provided the audio or visual alarm capability remains functional.

The system is designed to facilitate the maintenance and repair of malfunctioning components with minimum occupational radiation exposure.

4.4.3.5.2 System Description

There are ten (10) loose parts monitoring sensors (accelerometers) located in pairs to provide for sensor redundancy. Sensors are provided at the reactor vessel head lug, the reactor vessel bottom, and at each steam generator primary and secondary side.

Instrumentation channel components (including cabling and preamplifiers) associated with the sensors at each location are physically separated up to a point in the plant that is always accessible for maintenance during full power operation.

5.2.5.3.1.2 Reactor Coolant Pump Seals

Charging flow is directed to the RCP via a seal water injection filter. It enters the pumps at a point between the labyrinth seals and the No. 1 face seal. Here the flow splits and a portion (normally about 5 gpm) enters the reactor coolant system via the labyrinth seals and thermal barrier cooler cavity. The remainder of the flow (normally about 3 gpm) flows up the pump shaft cooling the lower bearing, and leaves the pump via the No. 1 seal where its pressure is reduced to about 25-30 psig and its temperature is increased from 130°F to about 136°F. The labyrinth flows (15 gpm total for three RCP) flow to a common manifold and then via a filter (seal water filter) through the seal water heat exchanger (where the temperature is reduced to about 130°F) to the volume control tank.

The leakoff system between No. 2 and No. 3 seals is considered to be part of the RCS. The leakoff system collects leakage passed by the No. 2 seal, provides a constant backpressure on the No. 2 seal and constant pressure on the No. 3 seal. A standpipe is provided to give a constant backpressure during normal operation. The first outlet from the standpipe is orificed to permit normal No. 2 seal leakage to flow to the reactor coolant drain tank; excessive No. 2 leakage will result in a rise in the standpipe level and eventual overflow to the reactor coolant drain tank via a second overflow connection.

The normal No. 2 seal flow is 3 gph/pump. This is the value specified in the RCP Equipment Specification.

Level instrumentation on the standpipes is provided to alert the operator to abnormal conditions. The standpipe consists of a pipe with an orificed overflow at the mid-point, a normally closed drain (for service) at the bottom, and a free flowing overflow at the top. Normal No. 2 seal leakage will flow freely out the mid-point overflow. Excessive leakage will "back-up" in the standpipe until it overflows out the top. A level switch in the upper

standpipe actuates an annunciator indicating excessive flow. A level switch in the lower standpipe causes annunciation of the opposite condition which could result in undesirable dry operation of the No. 3 seal.

5.2.5.3.1.3 Reactor Vessel Flange Leakoff

The reactor vessel flange and head is sealed by two metallic O-rings. To facilitate leakage detection, a leakoff connection is placed beyond the outer O-ring. Piping and associated valving is provided to direct any leakage to the reactor coolant drain tank.

During normal operation, it is anticipated that the leakage will be negligible since it is specified in the Reactor Vessel Equipment Specification that there is to be zero leakage past the outer O-ring under normal operating and transient conditions.

A temperature detector will indicate leakage by a high temperature alarm. The operator is further alerted by the associated increase in drain tank water temperature and eventually the change in tank level.

5.2.5.3.2 Paths Directed To Pressurizer Relief Tank (PRT)

The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharges from smaller relief valves located inside the containment are also piped to the PRT. During normal operation, leakage from the pressurizer safety valves, the pressurizer relief valves, or the CVCS letdown station relief valve is piped to the PRT.

During normal operation, the leakage to the PRT is expected to be negligible since the valves are designed for essentially zero leakage at the normal system operating pressure, as specified in the respective valve equipment specifications.

For each valve, temperature detectors are provided in the discharge piping to alert the operator to possible leakage.

The rate of increase of the water temperature in the PRT and the level change will indicate to the operator the magnitude of the leakage. In the event of excessive leakage into an interconnecting system causing lifting of the local relief valves, the operator would again be alerted to the situation by a rising tank water temperature.

5.2.5.3.3 Releases to the Containment Environment

The main contributors of leakage to the containment environment may be listed as follows:

a) Valve stem leakage

b) RCP seal No. 3 leakage

c) Flange leakages

TABLE 5.2.5-1 (Cont'd)

LEAKAGE TO INTERCONNECTION SYSTEMS

SYSTEM	DISCUSSION
SIS High Head Pump Injection Lines	In the event of leakage past two check valves and a normally closed motorized gate valve in the cold leg lines, and one check valve and a normally closed globe valve in the hot leg lines, pressure relief will take place to the PRT via the relief valve in the SIS test line.
SIS Accumulator Connections	Provisions have been made to check the leak tightness of the accumulator check valves. The implications of leakage past these valves are discussed in Section 6.2 of the FSAR.

CHAPTER 6
ENGINEERED SAFETY FEATURES

LIST OF FIGURES

CHAPTER 6
ENGINEERED SAFETY FEATURES

LIST OF FIGURES (Cont'd)

CHAPTER 6
ENGINEERED SAFETY FEATURES

LIST OF FIGURES (Cont'd)

CHAPTER 6
ENGINEERED SAFETY FEATURES

LIST OF FIGURES (Cont'd)

6.0 ENGINEERED SAFETY FEATURES

Eight types of engineered safety features are included in the design of the HBR 2 facility to mitigate the consequences of a postulated accident in spite of the fact that these accidents are very unlikely. These safety features are:

a) The Safety Injection (SI) System accumulators and pumps, which inject borated water into each coolant loop of the Reactor Coolant System (RCS). This system limits damage to the core and limits the energy released into the containment following a loss-of-coolant accident (LOCA).

b) The Containment Spray System, which is used to reduce containment pressure and to wash down iodine into the containment sump.

c) The air recirculation coolers, which reduce containment pressure following a LOCA.

d) The Post-Accident Venting System, through which hydrogen-free air may be admitted and hydrogen-bearing gases exhausted from the containment.

e) A steel-lined concrete containment structure described herein, with continuously pressurized penetrations and testable liner welds, which form a virtually leaktight barrier to the escape of fission products should a LOCA occur.

f) An Isolation Valve Seal Water System, which creates a leak tight seal in all pipes which could communicate with the atmosphere inside the containment following a LOCA.

g) A Containment Penetration Pressurization System, which provides a means to pressurize the containment penetrations above the maximum post-incident pressure.

h) A reactor coolant gas vent system which vents non-condensible gases from the reactor vessel head and the pressurizer steam space during post-accident situations. 3

The air recirculation system consists of four air handling units, each including air operated inlet louvers, roughing filters, cooling coils, fan and drive motor, duct distribution system, instrumentation, and controls. The units are located on the operating floor adjacent to the containment wall.

Each fan is designed to supply at least 65,000 cfm at design basis accident (DBA) conditions at approximately 20 in. s.p., 263°F, 0.162 lb/ft^3 density. 8
The fans are direct driven centrifugal type. Cooling coils are plate fin-tube type. Each air handling unit is capable of removing 40 x 10^6 Btu/hr from the containment atmosphere under DBA conditions. 750 gpm of service (cooling) water is supplied to each unit. The design maximum cooling water inlet temperature is 95°F which results in a maximum outlet temperature of 195°F under DBA conditions.

A gravity operated damper in the fan discharge isolates any inactive air handling unit from the duct distribution system. The damper opens automatically when the fan is started. Duct work distributes the cooled air to the various containment compartments and areas. The flow sequence through each air handling unit is as follows: roughing filter, inlet damper, cooling coils, fan, outlet louver, and discharge header for normal flow. For accident flow path, the inlet damper is closed and the flow enters the unit through a butterfly valve to the cooling coils.

Individual system components and their supports meet the requirement for Class I (Seismic) structures (Section 3.7) and each component is mounted to isolate it from fan vibration.

Actuation Provisions - The inlet dampers used to route air flow through the operating units have only two positions, full open or full closed. These louvers are air operated and spring loaded. Upon loss of control signal or control air, the spring actuates the louver to the closed position (fail-safe operation). The inlet butterfly valves used to route air flow through the operating units for accident conditions have only two positions, full close or full open. These valves are air operated and spring loaded. Upon loss of control signal or control air, the spring actuates the valve in the open position (fail-safe operation).

Redundant electrically operated three-way solenoid valves are used at each damper and butterfly valve to control the instrument air supply (control air). These valves are arranged so that failure of a single solenoid valve to respond to the accident signal will not prevent actuation of the louver or valve to the accident position (fail-safe operation). 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.

The fans are part of the ESF and either all four, or at least two of four fans will start after an accident, depending on the availability of emergency power (refer to Section 8.3).

Overload protection for the fan motors is provided at the switchgear by overcurrent trip devices in the motor feeder breakers. The breakers can be operated from the Control Room and can be reclosed from the Control Room following a motor overload trip.

Flow switches in the system, operating both normally and post-accident, indicate whether air is circulating in accordance with the design arrangement. Low flow alarms are provided in the Control Room.

Temperature elements (RTD's) are installed on the inlet and outlet (air side) of each fan cooler unit to provide data for monitoring cooling performance.

Flow Distribution and Flow Characteristics - The location of the distribution ductwork outlets, with reference to the location of the air handling unit returns inlets, ensures that the air will be directed to all areas requiring ventilation before returning to the units. The distribution system is represented schematically by the Ventilation Systems Flow Diagram, Figure 9.4.1-2.

The air discharged inside the reactor coolant loop shield walls will circulate and rise above the operating floor through openings around the SG and return to the air handling unit inlets. The temperature of this air will be essentially the ambient existing in the containment vessel.

The steam-air mixture from the containment entering the fan-cooler units during the accident will be at approximately 263°F and have a density of 0.162 lb per cubic foot. Part of the water vapor will condense on the cooling coils, and the air leaving the coils will be saturated at a temperature slightly below 263°F.

The fluid will remain in this condition as it flows into the fan, but will pick up some sensible heat from the fan and fan motor before flowing into the distribution header. This sensible heat will increase the dry-bulb temperature slightly above 263°F and will decrease the relative humidity slightly below 100 percent.

Cooling Water for the Fan Cooler Units - The cooling water requirements for all four fan cooling units during a major loss of primary coolant accident and recovery are supplied by two of the four service water pumps and one of the two service water booster pumps. The service water system is described in Section 9.2.1.

The cooling water discharges from the cooling coils to the discharge canal and is monitored for radioactivity by routing a small bypass flow from each unit through a common radiation monitor. Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil. The service water system is pressurized inside the containment, but the pressure in certain portions will be below the containment design pressure of 42 psig. However, since the cooling coils and service water lines form a closed system inside the containment, no contaminated leakage is expected into these units. Isolation valves on the inlet and discharge of each fan cooler are located outside the containment and may be used to isolate individual fan cooler units in the event that radioactivity is detected by the radiation monitor.

Local flow and temperature indication is provided outside containment for service water flow from each cooling unit.

Local temperature elements (RTD's) on the water inlets and outlets and a local pressure differential indicator and pressure gauges are installed on each fan cooler unit to provide data for monitoring cooling coil performance.

<u>Environmental Protection</u> - All system control and instrumentation devices required for containment accident conditions are located to minimize the danger of control loss due to missile damage. Flow switches in the ductwork system, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the Control Room.

6.2.2.4 Tests and Inspection

6.2.2.4.1 Containment Spray System

All components of the CSS can be inspected periodically to demonstrate system readiness.

The pressure containing systems are inspected for leaks from pumps seals, valves packing, flanged joints, and safety valves during system testing. During the operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks. The inservice inspection program for HBR 2, is discussed in Section 3.9.

Component Testing - All active components in the CSS were tested both in pre-operational performance test in the manufacturer's shop and in-place testing after installation.

The containment spray pumps can be tested singly by opening the valves in the miniflow line. Each pump in turn can be started by operator action and checked for flow establishment. The spray injection valves can be tested with the pumps shut down.

The spray additive tank valves can be opened periodically for testing. The contents of the tank will be periodically sampled to determine that the proper solution is present.

Initially, the containment spray nozzle availability was tested by blowing smoke through the nozzles and observing the flow through the various nozzles in the containment.

During these tests the equipment was visually inspected for leaks. Leaking seals, packing, or flanges were tightened to eliminate the leak. Valves and pumps have been operated and inspected after any maintenance to ensure proper operation.

System Testing - Permanent test lines for all containment spray loops were located so that the system, up to the isolation valves at the spray header, can be tested. These isolation valves can be checked separately.

The air test lines, for checking initially the spray nozzles, connect downstream of the isolation valves. Air flow through the nozzles is monitored by the use of hot air and infrared thermography.

During the initial pre-operational tests of the spray system, the flow bypass through the spray eductors was checked. This initial and all subsequent system tests were made with the spray additive tank isolation valves closed.

Operational Sequence Testing - The functional test of the SIS described in Section 6.3.4 demonstrated proper transfer to the emergency DG power source in the event of loss of power. A test signal simulating the containment spray signal has been used to demonstrate operation of the spray system up to the isolation valves on the pump discharge.

6.2.2.4.2 Containment Air Recirculation Cooling System

Access is available for visual inspection of the containment air recirculation system components including fans, cooling coils, louvers, and ductwork.

The air recirculation cooling units, and the service water pumps and booster pumps which supply the cooling units, are in operation on an essentially continuous schedule during plant operation, and no additional periodic tests are required.

Component Testing - The roughing filters used in the containment fan cooler system are specified to operate in the normal containment environment. The filters are subjected to standard manufacturer's efficiency and production tests prior to shipment.

Reactor Containment Fan Cooler Motor Unit Tests - The testing program has been completed on the effects of radiation on the WF-8AC "Thermalastic" (Westinghouse Electric Corporation Trademark) epoxy insulation system used in the reactor containment fan cooler motor. Tests description and results are presented in Reference 6.2.2-10.

Fan Cooler Motor Insulation Irradiation Testing - This testing program is an extension of the work reported in Reference 6.2.2-11.

Irradiation of form wound motor coil sections was accomplished up to exposure levels exceeding that calculated for the design basis LOCA. Three coil samples received the following treatment sequence: Irradiation, high-potential test, vibration test, high-potential test, and breakdown voltage test. Nine coil samples received an alternate treatment sequence: Thermal aging, high-potential test, irradiation, high-potential test, vibration test. (Six of nine coil samples), high-potential test and breakdown voltage test.

All coil samples passed the high potential tests. The breakdown voltage levels of all coils were well in excess of those required by the design, and clearly indicate that the reactor containment fan cooler motor insulation system will perform satisfactorily following exposure to the radiation levels calculated for the DBA.

Reactor Containment Fan Cooler Motor Lubricant Irradiation Testing - The lubricant used in the containment fan cooler motors is qualified for its applicable service. Testing documentation is located in the EQ Central File.

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8. Isolated lines between the containment and the second outside isolation barrier (valve or closed system) were designed to the same seismic criteria as the containment vessel, and were assumed to be an extension of containment.

9. The first outside isolation valve is located as close to the containment as possible unless a more remote location was dictated by equipment isolation requirements.

The six classes of piping penetrations are:

1. Class I (Outgoing Lines, RCS) - Normally operating outgoing lines connected to the RCS are provided with at least two automatic trip valves in series located outside the containment. Automatic seal water injection is provided for lines in this classification.

An exception to the general classification is the residual heat removal (RHR) loop outlet line, which has two automatic barriers--one established by a normally closed missile protected valve inside containment, the other by the closed RHR loop outside containment.

2. Class 2 (Outgoing Lines) - Normally-operating outgoing lines not connected to the RCS, and not missile-protected, or which can otherwise communicate with the containment atmosphere following an accident, were provided at a minimum with two automatic trip valves in series outside containment. Automatic seal water injection is provided for lines in this classification. Most of these lines are not vital to plant operation following an accident.

3. Class 3 (Incoming Lines) - Incoming lines connected to open systems outside containment, and not missile-protected or which can otherwise communicate with the containment atmosphere following an accident are provided with one of the following arrangements outside containment:

a. Two automatic trip valves in series, with automatic seal water injection. This arrangement was provided for lines which are not necessary to plant operation after an accident.

b. Two manual isolation valves in series, with manual seal water injection. This arrangement was provided for lines which remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile-protected, or which can otherwise communicate with the containment atmosphere, are provided at a minimum with one check valve or normally closed isolation valve located either inside or outside containment with the exception that the containment isolation valves for S. I. cold leg injection are normally open since this is the required safety position. The closed piping system outside containment provides the necessary isolation redundancy. Most lines in this category were provided with additional isolation valves which satisfy particular systems or safeguards requirements. Seal water injection was provided for certain lines in this category.

4. Class 4 (Missile Protected) - Normally operated incoming and outgoing lines which penetrate the containment and are connected to closed systems inside the containment and protected from missiles throughout their length. These lines are provided with at least one manual isolation valve located outside the containment. Seal water injection was not required for this class of penetration.

5. Class 5 (Normally Closed Lines Penetrating the Containment) - Lines which penetrate the containment and which can be opened to the containment atmosphere but which are normally closed during reactor operation were provided with two isolation valves in series or one isolation valve and one blind flange. One valve or flange is located inside and the second valve or flange is located outside the containment.

6. Class 6 (Special Service) - There are a number of special groups of penetrating lines and containment access openings. These are discussed below.

Each ventilation purge duct penetration was provided with two tight-closing butterfly valves, which are closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration. The space betweeen valves is pressurized by air from the Penetration Pressurization System (PPS), Section 6.9, whenever they are closed.

The containment pressure and vacuum relief lines were similarly protected with two tight closing butterfly valves in series, one inside and one outside the containmemt. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal with the intravalve spaces pressurized by air from the PPS.

The equipment access hatch is a bolted, gasketed closure which is sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to assure that one door was closed at all times. Each air lock door and the equipment closure is provided with double gaskets to permit pressurization betwen the gaskets by the PPS, Section 6.9.

The fuel transfer tube penetration inside the containment, Figure 3.8.1-16, was designed to present a missile-protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the containment. The penetration closure was treated in a manner similar to the equipment access hatch. The inside closure is a blind flange which contains two gaskets. A positive pressure is maintained between these gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube was not pressurized. Seal water injection was not required for this penetration.

The containment radiation monitor inlet and outlet lines communicate with the containment atmosphere at all times (normally filled with air or vapor). In an accident condition the space between the two containment isolation valves in each line is sealed by pressurizing with air from the PPS. The air is introduced into each space at approximately 2 psi above the containment design pressure through a separate line from the PPS. Parallel (redundant) fail open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air.

A flow limiting orifice in each injection line prevents excessive air consumption if one of these valves spuriously fails to open, or if one of the containment isolation valves fails to respond to the "trip" signal.

Figures 6.2.4-1 through 6.2.4-20 show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure modes, the application of "trip" (containment isolation) signals, relative location of the valves with respect to missile barriers, and the boundaries of Seismic Class I design lines. Figure 6.2.4-21 defines the nomenclature and symbols used.

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 6.2.4-1. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown, and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by containment isolation signal, and the fluid carried by the line.

Containment isolation valves were provided with actuation and control equipment appropriate to the valve type. For example, air-operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation assured by the control devices in the instrument air supply to the valve. Motor-operated gate valves are capable of being supplied from reliable onsite emergency power as well as from their normal power source.

The automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "non-essential"* process lines penetrating the containment. This was defined as "Phase A" isolation, and the trip valves were designated by the letter "T" in the isolation diagrams (Figures 6.2.4-1 through 6.2.4-20). This signal also initiates automatic seal water injection. The second, or "Phase B", containment isolation signal was derived upon actuation of the containment spray system, and trips the automatic isolation valves in the so-called "essential"* process lines penetrating the containment. These trip valves were designated by the letter "P" in the isolation diagrams.

A manual containment isolation signal can be generated from the Control Room. This signal performs the same functions as the automatically derived "T" signal, i.e. "Phase A" isolation and automatic seal water injection.

Non-automatic isolation valves, i.e., remote stop valves and manual valves, were used in lines which must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

*"Non-essential" process lines are defined as those which do not increase the potential for damage to in-containment equipment when isolated. "Essential" process lines are those providing cooling water and seal water flow through the reactor coolant pumps. These services should not be interrupted unless absolutely necessary while the reactor coolant pumps are operating.

Standard closing times available with commercial valve modes were adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately two seconds. The typical closing time available for large motor-operated gate valves was ten seconds. (Ref. GID/90-181/00 RCI Appendix B)

The large butterfly valves used to isolate the containment ventilation purge ducts were equipped with air-cylinder operators, with spring returns capable of closing the valves in two seconds. These valves fail to the closed position on loss of control signal or instrument air.

The following types of isolation valves were employed outside the containment:

1. Diaphragm valves (Saunders Patent)
2. Globe valves
3. Double disk gate valves
4. Regular gate valves, and
5. Butterfly valves.

Diaphragm valves were used in 3 in. diameter lines and smaller, with low pressure, low temperature service requirements (200 psig, 200°F or less). Two remote operated valves in series were used in non-missile protected lines going out of the containment. Seal water is injected between the valves at a pressure slightly higher than the containment design pressure, to ensure that any leakage past the diaphragms of the closed valves will be seal water and not containment atmosphere. Lines so protected are isolated automatically by the containment isolation signal, which also initiates the seal water flow. A number of penetrating lines are isolated by one manual or remote operated diaphragm valve. These lines were missile-protected inside the containment, and some were connected to external systems which have sufficient capacity for inflow to the containment.

Two manual or air operated globe valves in series were used to isolate lines 2 in. and smaller with design pressure or temperature greater than 200 psig or 200°F, respectively. Seal water is injected between the valves, which were installed so that the zone between the seat or stem packing will be seal water and not containment atmosphere.

Liquid carrying lines 2 in. and larger with design pressure or temperature exceeding 200 psig or 200°F were isolated by one manual or remote operated double disk gate valve. This type of valve provides two barriers against leakage of containment atmosphere. Seal water is injected into the valve body on closing, to ensure that any leakage past the valve disks or stem will be seal water and not containment atmosphere.

Regular gate and butterfly valves were used outside the containment to isolate penetrating lines that are missile protected and connected to closed systems inside the containment.

Isolation valves with packed stems were provided with steam leakoffs if all of the following operating conditions were satisfied with the exception of those valves which have been live loaded and have had leakoff lines capped:

a) Line size is 2 in. or larger

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,

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

3 h) Reactor vessel level instrumentation system lines.

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 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.

Containment liner weld channels were installed on all liner welds to provide the means for a sensitive leak rate test to determine liner weld leakage (L_c). However, the liner weld leakage is no longer determined and the integrity of the containment welds was verified by periodic integrated leakage rate testing.

Containment isolation valves were individually tested prior to the preoperational leak rate tests to assure proper seating. The design of the lines which penetrate the containment boundary provide isolation valves and additional positive means for limiting the leakage (L_{iso}) which can occur from the containment atmosphere through these lines in the post-accident condition. Table 6.2.4-1 lists each fluid line which penetrates the containment wall and indicates the additional positive barriers which will minimize leakage through these lines from the containment following an accident. These positive barriers include:

a) Injection of IVSW System water at a pressure greater than accident pressure between the seats and stem packing of the globe and double disc types of isolation valves and into piping between closed diaphragm type isolation valves.

b) Injection of PPS air into the spaces between isolation valves in lines connecting directly with the containment atmosphere.

Other lines are all located outside the missile barrier and are connected to closed systems within the containment, or are part of a system with design pressure greater than the design pressure of the containment. Therefore, the isolation valve arrangement and these positive barriers will assure minimal leakage after a DBA through these potential leak paths.

No leakage was expected through the liner plates (L_L). However, any liner plate leakage will be measured as part of the preoperational integrated leakage rate test. The containment liner has insulation from the area of the "spring line" to the base mat. This polyvinyl chloride (PVC) foam insulation has a sheet stainless steel outer covering. Any physical damage to this insulation and thus to the underneath liner would be readily observable.

Following the preoperational tests, periodic inspection of the containment wall was conducted to ensure that no physical damage to the liner has occurred. Evidence of damage would be examined to determine the necessary methods for assuring that the liner plate(s) in the affected area will not leak at containment design conditions. Therefore, no periodic leak rate testing of the liner plates is required unless physical damage was evident.

The preoperational integrated leak rate test was conducted with containment atmosphere at 42 psig and 90^{o}F. The corresponding test leakage rate limit was 0.0761 percent of the containment free volume per day. The total integrated leakage rate of the containment vessel was described by the following equation:

$$L_{pm} = L_{pen} + L_c + L_{iso} + L_L \qquad (1)$$

The actual integrated leak rate (L_{pm}) measurement was preceded by individual leak testing of containment isolation valves and penetrations to determine and correct, if necessary, leakage paths which might be present. The integrated leak rate (L_{pm}) was determined during a twenty-four hour test period by taking hourly readings of containment internal pressure, temperature, and dew point temperature. These parameters were then introduced into an equation derived from the ideal gas law to establish the weight of air lost per unit time. The limit on weight lost per unit time at the test conditions of 42 psig and 90°F is 446 lb/day.

The error (p) associated with the integrated leak rate measurement must be considered to evaluate the results of the test. Therefore, the following relationship must be satisfied to assure that the containment vessel meets its design criterion:

$$L_{pm} + \sigma_p \leq 0.0761 \text{ percent per day at 42 psig and } 90^{\circ}\text{F} \qquad (2)$$

The integrated leak rate test (Test Case I) was conducted with the containment isolation valves in their post-accident condition without utilization of the IVSW System, and without PPS pressure to the penetrations. Pressure buildup was observed to evaluate leakage from the containment. This measurement of the integrated leakage rate along with the preoperational sensitive leakage rate tests provides a basis for evaluation of operational sensitive leak rate tests, to confirm periodically that the containment leakage rate is within the design limit.

After the twenty-four hour integrated leakage rate test, a controlled leakage rate equal to 0.0761 percent per 24 hr was superimposed on the containment for 12 hr. This procedure is intended to validate the method and instrumentation used for the integrated leak rate test.

Leakage through containment penetrations (L_{pen-2}) to the outside was measured by the first phase of the preoperational sensitive leak rate test following the integrated leak rate test.

The containment pressure was then reduced to 41 psig and the penetrations were pressurized to 42 psig (Test Case II). This arrangement prevents inleakage from the containment atmosphere to the test channel volume during the sensitive leak test. Leakage to the outside was measured using the flow instrumentation of the PPS, and was subject to the measurement error, σ_{pen-2}.

The containment pressure was then reduced to 0 psig and the penetrations were pressurized to 42 psig (Test Case III). The leakages from these volumes, L_{pen-3}, respectively, were then measured. These leakages were subject to measurement error, σ_{pen-3}, and represents the total leakage from these volumes to both the containment interior and to the outside environment.

Therefore, these sensitive leak rate tests permitted determination of leakage through the outer and inner barriers of the penetrations.

The integrated leak rate tests are summarized as follows:

a) Test Case I

Condition: Containment at 42 psig, penetrations at 0 psig.

$L_{pen-1} + \sigma_{pen-1}$ describes penetration inleakage.

b) Test Case II

Conditions: Containment at 41 psig, penetrations at 42 psig.

$L_{pen-2} + \sigma_{pen-2}$ describes penetration leakage to the outside assuming no leakage to the containment.

c) Test Case III

Conditions: Containment at 0 psig, penetrations at 42 psig.

$L_{pen-3} + \sigma_{pen-3}$ describes penetration leakage.

Leakage to outside from the penetrations was assumed to be equal for Test Cases II and III since the penetrations were at 42 psig.

A value for leakage through liner plates and isolation valves ($L_L + L_{iso}$) can be obtained from the preoperational integrated leak rate and sensitive leak rate tests by substituting the measured integrated leakage into Equation. (1)

$$L_{pm} + \sigma_p = L_{pen} + L_c + L_{iso} + L_L \qquad (3)$$

where $L_{pen} + L_c$ was obtained from the sensitive leak rate tests (Test Case II). Therefore:

$$L_{iso} + L_L = L_{pm} + \sigma_p - \left[(L_{pen-2} \pm \sigma_{pen-2}) + (L_{c-2} \pm \sigma_{c-2}) \right] \qquad (4)$$

Therefore, the sum of leakages through isolation valves and liner plates ($L_{iso} + L_L$) can be estimated during the preoperational leak rate testing.

Periodic sensitive leak rate tests are performed on the penetrations and isolation valve and seals receiving PPS pressure to assure that the leakage from these most probable leak paths has not increased sufficiently since the preoperational testing to result in a containment leak rate exceeding 0.1 percent of the containment volume at design conditions. This periodic sensitive leak rate test (Test Case IV) was performed with the containment at zero psig and the penetrations, isolation valves and seals at 42 psig, which were the same test conditions at Test Case III discussed above. Leakage from the weld channels and penetrations, L_{c-4} and L_{pen-4}, and their respective measurement errors, σ_{c-4} and σ_{pen-4}, can be represented by:

$L_{c-4} + \sigma_{c-4}$ describes weld channel leakage.

$L_{pen-4} + \sigma_{pen-4}$ describes penetration leakage.

The current status of the containment leakage can then be established by comparing the results of the periodic operational sensitive leak test with preoperational leak rate data. From preoperational sensitive leak rate testing, $L_{pen-3} + L_{c-3}$ was measured. These measurements include leakage from the penetrations to the inside and outside of the containment, and leakage from the isolation valves and seals. Any increase in the periodic sensitive leak rates $L_{pen-4} + L_{c-4}$ over preoperational sensitive leak rates was assumed to be leakage to the outside. This assumption is conservative in that any leakage increase detected in periodic sensitive leak rate testing might consist of increased leakage to the containment interior as well as increased leakage to the environment. The periodic sensitive leak rates, $L_{pen-4} + L_{c-4}$, were measured to establish the current condition of the containment.

Therefore, the total containment leakage (L_p) as a result of increased leakage detected by periodic sensitive leak rate testing would be:

$$L_{pm} + \sigma_p + \left[(L_{pen-4} + \sigma_{pen-4}) - (L_{pen-3} \pm \sigma_{pen-3}) \right] \quad (5)$$

$$\left[(L_{c-4} + \sigma_{c-4}) - (L_{c-3} \pm \sigma_{c-3}) \right] \leq 0.0761 \text{ percent}$$

of the containment volume at 42 psig and 90^{o}F.

Equation (5) represents the criterion which must be satisfied as a result of the periodic sensitive leak rate tests to verify that the containment leak rate does not exceed its design limit. The equation takes into account the preoperational integrated leak rate and the relative changes in sensitive leak rates as determined by periodic sensitive leakage measurements.

Initial failure of the sensitive leak rate test to verify this relationship will result in efforts to reduce leakage from the penetrations, and isolation valves and seals so that a sensitive leak rate retest will demonstrate compliance with the design limit.

Periodic sensitive leak rate testing and preoperational leak rate testing discussed above describe a reasonable approach to assuring that the containment leakage is maintained below design limits during the life of the plant. Periodic sensitive leak testing provides a very accurate method of monitoring changes in the leakage characteristics of the containment. An integrated leak rate test was performed if major maintenance or modification to the containment was made.

In-service testing of pumps and valves to assure operability under design loading conditions, and preoperational testing to assure operability when subjected to dynamic loading conditions associated with system transient conditions, are described in Section 3.9.6. Additional in-service inspection requirements are discussed in Section 3.9.

6.2.4.4 Gas Analyzer Isolation Valves

The control circuitry associated with the containment isolation valves in the Gas Analyzer Sample Line from the Pressurizer Relief Tank (RC-516 and RC-553) and the containment isolation valves in the Gas Analyzer Sample Line from the Reactor Coolant Drain Tank (WD-1789 and WD-1794) is such that the valves will close in any and all of the following cases:

a) "CLOSE" command from the Gas Analyzer Panel

b) Containment Phase A Isolation Signal, and

c) Loss of Power.

If the valves have closed as a result of a "CLOSE" command from the Gas Analyzer Panel, they can be reopened by an "OPEN" command from the Gas Analyzer Panel as long as there has been no Containment Phase A Isolation Signal or Loss of Power. In the latter two cases, the valves can only be reopened by first resetting the Phase A Isolation Signal and/or reestablishing power to the circuitry.

The valves will continue to remain closed until the corresponding Isolation Reset Pushbuttons (one per valve) are depressed. Once this is accomplished, the valves will reopen. These valves were initially resettable in a ganged fashion, but are presently resettable on a valve-by-valve basis. These valve systems have been modified such that the resetting of containment isolation will not result in automatic reopening.

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS AND VALVING

C.I.S. - Containment Isolation Signal
A - Automatic Seal Water Injection
M - Manual Seal Water Injection
FC - Fail Closed
Dia. - Diaphragm Valve
DDV - Double Disc Gate Valve
C.S. - Closed System

Hot - >200°F
Cold - <200°F

RCS - Reactor Coolant System
ACS - Auxiliary Coolant System
WDS - Waste Disposal System
SIS - Safety Injection System
SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
1	Pressurizer	-1	A	Dia.	Air	Yes	Open	Open	Closed	FC	T	A	No	G	Cold	
	Relief Tank		B	Dia.	Air	Yes	Open	Open	Closed	FC	T	A	No	G	Cold	
	Sample - RCS		C	Globe	Sol.	No	Closed	Closed	Closed	Closed	No	-	No	G	Cold	Opened intermittently by automatic analyzer for gas sample.
2	Pressurizer	-1	A	Check	-	No	Closed	Closed	Closed	-	No	-	No	G	Cold	
	Relief Tank		B	Dia.	Air	Yes	Open	Open	Closed	FC	T	-	No	G	Cold	
	Nitrogen Supply-RCS		C	Press. Reg.	Self Con- tained	No	Closed	Closed	Closed	FC	No	-	No	G	Cold	Regulator set for 3 psig
3	Pressurizer	-1	A	Dia.	Air	Yes	Closed	Closed	Closed	FC	T	A	No	W	Cold	
	Relief Tank Makeup-RCS		B	Dia.	Air	Yes	Closed	Closed	Closed	FC	T	A	No	W	Cold	
4	Primary Sys-	-2	A	Dia.	Air	Yes	Open	Open	Closed	FC	T	A	No	G	Cold	
	tem Vent		B	Dia.	Air	Yes	Open	Open	Closed	FC	T	A	No	G	Cold	
	Header		C	Dia.	Man.	No	Closed	Closed	Closed	As is	No	-	No	G	Cold	
	Nitrogen Sup-		D	Check	-	No	Closed	Closed	Closed	-	No	-	No	G	Cold	
	ply Line-WDS		E	Press. Ref.	Self Contained	No	Open	Open	Closed	FC	No	-	No	G	Cold	Regulator set for 3 psig

C.I.S. - Containment Isolation Signal
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PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
5	Reactor Coolant Drain Tank	-2	A	Globe	Air	No[a]	Open	Open	Closed	FC	T	A	No	G	Cold	Opened intermittently by automatic analyzer for gas sample.
			B	Globe	Air	No[a]	Open	Open	Closed	FC	T	A	No	G	Cold	
			C	Globe	Sol.	No	Closed	Closed	Closed	FC	No	-	No	G	Cold	
6	Reactor Coolant Drain Tank Pump Discharge Line WDS	-2	A	Dia.	Man.	No	Closed	Closed	Closed	As is	No	-	No	W	Cold	
			B	Dia.	Air	No[a]	Open	Open	Closed	FC	T	A	No	W	Cold	
			C	Dia.	Air	No[a]	Open	Open	Closed	FC	T	A	No	W	Cold	
7 8 9	Main Stream Headers (Sec. Sys.)	-3	A	Swing-Disc Stop Valve	Air Pis-ton	Yes	Open	Closed	Open*	Shut	No*	-	Yes*	G	Hot	*Automatic isolation for stream line break.
			B	Gate	Mot.	Yes	Closed	Closed	Closed	As is	No	-	No	G	Hot	
			C	Check	-	No	Open	Open	Open	As is	No	-	Yes	G	Hot	
			D	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	G	Hot	
			E	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	G	Hot	
			F	Globe	Mot.	Yes	Closed	Closed	Open	As is	No	-	Maybe	G	Hot	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Steam generator secondary side is a missile protected closed system.

[a]Indicated at the Waste Disposal System (WDS) Panel

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TABLE 6.2.4-1 (Cont'd)

C.I.S. - Containment Isolation Signal
A - Automatic Seal Water Injection
M - Manual Seal Water Injection
FC - Failed Closed
Dia. - Diaphragm Valve
DDV - Double Dis Gate Valve
C.S. - Closed System

Hot - >200°F
Cold - <200°F

RCS - Reactor Coolant System
ACS - Auxiliary Coolant System
WDS - Waste Disposal System
SIS - Safety Injection System
SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
10	Feed-	-3	A	Check	-	No	Open	Open	Open*	As is	No	-	Yes*	W	Hot	*Isolated for steam
11	water		B	Gate	Man.	No	Open	Open	Open*	As is	No	-	Yes*	W	Hot	line break.
12	Headers		C	Gate	Man.	No	Open	Open	Open*	As is	No	-	Yes*	W	Hot/Cold	
	(Sec. Sys.)		D	Check	-	No	Open	Open	Open*	As is	No	-	Yes*	W	Cold	This valve is in emergency feedwater line. Steam generator secondary side is a missile protected closed system.
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	
			E	Globe	Man.	No	Closed	Open	Closed	As is	No	-	No	W	Hot	
			F	Globe	Man.	No	Open	Open	Closed	As is	No	-	No	W	Cold	
13	Steam Gen.	-3	A	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	Hot	
14	Blowdown lines		B	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	Hot	
15	(Sec. Sys.)															
16	Residual	-4	A	DDV	Motor	Yes	Closed	Open*	Open	As is	No	-	Yes	W	Hot	*Open if cold
	Heat Removal		B	DDV	Motor	Yes	Open	Closed	Open*	As is	No	-	Yes*	W	Cold	*Closed during re-
	Loop Out-		C	DDV	Motor	Yes	Open	Closed	Open*	As is	No	-	Yes*	W	Cold	circulation phase.
	ACS/SIS		D	DDV	Motor	Yes	L.C.	Closed	Close**	As is	No	-	Yes	W	Hot/Cold	**Open during initial
			E	Gate	Motor	No	Open	Open	Opens	As is	No	-	Yes	W	Cold	recirculation phase.
			F	Globe	Man.	No	Open	Open	Open	As is	No	-	Yes	W	Hot/Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	RHR System

TABLE 6.2.4-1 (Cont'd)

C.I.S. - Containment Isolation Signal
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M - Manual Seal Water Injection
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Dia. - Diaphragm Valve
DDV - Double Disc Gate Valve
C.S. - Closed System

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SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
17	Residual Heat Removal Loop In - ACS/SIS	-5	A	Check	-	No	Closed	Open*	Open	As is	No	-	Yes	W	Hot/Cold	*If in cold S/D Opens on safety injection signal.
			B	Gate	Mot.	Yes	Closed	Open*	Open	As is	No	-	Yes	W	Hot/Cold	
			C	Globe	Man.	No	Open	Open	Closed	As is	No	-	No	W	Hot	Line normally closed during power operation.
			D	Globe	Air	Yes	Closed	Closed	Closed	As is	No	-	No	W	Hot	
			E	Globe	Man.	No	Open	Open	Open	As is	No	-	Yes	W	Hot	
			F	Butter-fly	Air	Yes	Closed	Closed**	Closed/Open	FC	No	-	Yes	W	Hot	**Open if in cold S/D
			G	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	Cold	*May be opened during accident for flow through containment spray pumps and high head safety injection pumps.
			H	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	W	Hot	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Residual heat removal loop.
18	Reactor Coolant Pump Cooling Water In - ACS	-6	A	Check	-	No	Open	Open	Closed	As is	No	-	No	W	Cold	
			B	DDV	Mot.	Yes	Open	Open	Closed*	As is	P	A	No	W	Cold	*If receive P signal
			C	DDV	Mot.	Yes	Open	Open	Closed*	As is	P		No	W	Cold	*If receive P signal
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Component cooling water loop.

TABLE 6.2.4-1 (Cont'd)

C.I.S. - Containment Isolation Signal
A - Automatic Seal Water Injection
M - Manual Seal Water Injection
FC - Fail Closed
Dia. - Diaphragm Valve
DDV - Double Disc Gate Valve
C.S. - Closed System

Hot - >200°F
Cold - <200°F

RCS - Reactor Coolant System
ACS - Auxiliary Coolant System
WDS - Waste Disposal System
SIS - Safety Injection System
SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
	Reactor Vessel		A	Check	-	No	Open	Open	Open	As Is	No	-	Yes	W	Cold	Isolators are designed for
	Instrumentation		B	Check	-	No	Open	Open	Open	As Is	No	-	Yes	W	Cold	the Reactor Coolant System
	System (RVLIS)		C	Isolator	-	Yes	-	-	-	-	-	-	Yes	W	Cold	Pressure of 2486 psig.
			D	Isolator	-	Yes	-	-	-	-	-	-	Yes	W	Cold	

3

TABLE 6.2.4-1 (Cont'd)

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CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

Hot - >200°F
Cold - <200°F

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
19	Reactor Coolant Pump Cooling Water Out - ACS (6" Line)	-6	A	DDV	Mot.	Yes	Open	Open	Closed*	As is	P	A	No	W	Cold	*If receive P signal
			B	Gate	Man.	No	Open	Open	Closed	As is	No	-	No	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Component cooling water loop.
20	Reactor Coolant Pump Cooling Water Out - ACS (3" Line)	-6	A	Gate	Mot.	Yes	Open	Open	Closed*	As is	P	A	No	W	Cold	*If receive P signal
			B	Gate	Mot.	Yes	Open	Open	Closed*	As is	P	A	No	W	Cold	
			C	Globe	Man.	No	Open	Open	Open	As is	No	-	No	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Component cooling water loop.
21	Excess Letdown Heat Exchanger Cooling Water In - ACS	-7	A	Check	-	No	Open	Open	Open	As is	No	-	No	W	Cold	
			B	Gate	Man.	No	Open	Open	Open	As is	No	-	No	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Component cooling water. Heat exchanger is a missile protected closed system inside containment.
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	

TABLE 6.2.4-1 (Cont'd)

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SIS - Safety Injection System
SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
22	Excess Letdown Heat Exchanger Cooling Water Out - ACS	-7	A	Gate	Air	Yes	Open	Open	Closed	FC	T	-	No	W	Cold	
			B	Gate	Man.	No	Open	Open	Open	As Is	No	-	No	W	Cold	
			C	Globe	Man.	No	Open	Open	Open	As Is	No	-	No	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Component cooling water loop. Heat exchanger is a missile protected closed system inside containment.
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	
23	Letdown Line CVCS	-8	A	Globe	Air	Yes	Open*	Closed	Closed	FC	T	-	No	W	Hot	Letdown orifice isolation valves. *One valve normally open, two closed.
			B	Globe	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	Hot	
			C	Globe	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	Hot	
			D	Globe	Man.	No	[a]Open	Open	Closed	As Is	No	-	No	W	Hot	Can be closed to
							[b]Closed	Open*	Closed	As Is	No	-	No	W	Hot	prevent loss of seal water if valve C fails to close tightly.
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Chemical and Volume Control System. *If in cold S/D

TABLE 6.2.4-1 (Cont'd)

C.I.S. - Containment Isolation Signal
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SIS - Safety Injection System
SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
24	Charging Line - CVCS	-8	A	Check	-	No	Open	Closed	Closed*	As is	No	-	No*	W	Cold	
			B	Globe	Man.	No	Open	Open	Closed*	As is	No	M	No*	W	Cold	*May be used for high
			C	Gate	Man.	No	Open	Open	Closed*	As is	No	M	No	W	Cold	pressure safety in-
			D	Globe	Man.	No	Closed	Closed	Closed*	As is	No	M	No*	W		jection.
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Chemical and Volume Control System
25	Reactor	-9	A	Check	-	No	Open	Open	Closed*	As is	No	-	Yes*	W	Cold	*Line is isolated after
26	Coolant Pump															pump is shutdown.
27	Seal Water		B	Needle	Man.	No	Throt.	Open	Closed*	As is	No	M	Yes*	W	Cold	
	Supply Lines		C	Globe	Man.	No	Open	Open	Closed*	As is	No	M	Yes*	W	Cold	
	- CVCS		D	Globe	Man.	No	Closed	Closed	Closed	As is	No	M	Yes*	W	Cold	
			E	Globe	Man.	No	Open	Open	Closed*	As is	No	M	Yes*	W	Cold	
			F	Gate	Man.	No	Closed	Closed	Closed	As is	No	M	Yes*	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Chemical and Volume Control System
28	Reactor	-9	A	DDV	Mot.	Yes	Open	Open	Closed*	As is	P	A	No	W	Cold	
	Coolant Pump		B	Dia.	Man.	No	Open	Open	Open*	As is	No	-	No	W	Cold	*If receive P signal
	Seal Water		C	Dia.	Man.	No	Open	Open	Open	As is	No	-	No	W	Cold	
	Return Line - CVCS															
			D	Dia.	Man.	No	Closed	Closed	Closed	As is	No	-	No	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Volume control tank.

TABLE 6.2.4-1 (Cont'd)

C.I.S. - Containment Isolation Signal
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Hot - >200°F
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ACS - Auxiliary Coolant System
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SIS - Safety Injection System
SS - Sampling System
CVCS - Chemical and Volume Control System
VENT - Ventilation System
PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
29	Reactor	-10	A	Globe	Air	Yes	Closed	Closed	Closed	FC	T	A	No	W	Hot	
30	Coolant Sys-		B	Globe	Air	Yes	Closed	Closed	Closed	FC	T	A	No	W	Hot	
31	tem Sample Lines - SS		C	Globe	Man.	No	Open	Closed	Open	As Is	No	-	No	W	Hot	Prevents loss of seal water if valve B fails.
32	Fuel Trans- fer Tube - FH	-10	A Flange	Blind	-	No	Closed	Closed	Closed	As Is	No	-	No	W	Cold	Double gasketed blind flange located in missile protected refueling canal.
			B	Gate	Man.	No	Closed	Closed	Closed	As Is	No	-	No	W	Cold	
33	Instrument	-11	A	Check	-	No	Open	Open	Closed	As Is	No	-	No	G	Cold	
	Air Header		B	Globe	Air	Yes	Open	Open	Closed	FC	T	-	No	G	Cold	
	(Sec. Sys.)		C	Globe	Man.	No	Open	Open	Open	As Is	No	-	No	G	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Instrument air supply is capable of providing inflow until line can be isolated, if line is ruptured inside containment. Any leakage will be toward containment.
34A	Nitrogen Sup-	-12	A	Globe	Man.	No	Closed	Closed	Open	As Is	No	-	Yes	G	Cold	Used to supply N_2 to opera-
34B	ply to Hydro-		B	Globe	Man.	No	Closed	Closed	Closed	As Is	No	-	No	G	Cold	tors on the post accident H_2
34C	gen Vent		C	Globe	Man.	No	Closed	Closed	Open	As Is	No	-	Yes	G	Cold	venting valves inside CV.
34D	System Valves															

TABLE 6.2.4-1 (Cont'd)

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PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
35	Containment Air Sample In - Rad. Mon.	-11	A	Dia.	Air	Yes	Open	Open	Closed*	FC	T	-	No*	G	Cold	*May be opened after accident for sample, when containment pressure is below 5 psig.
			B	Dia	Air	Yes	Open	Open	Closed*	FC	T	-	No*	G	Cold	
36	Containment Air Sample Out - Rad. Mon.	-11	A	Dia.	Air	Yes	Open	Open	Closed*	FC	T	-	No*	G	Cold	*May be opened after accident for sample, when containment pressure is below 5 psig.
			B	Dia.	Air	Yes	Open	Open	Closed*	FC	T	-	No*	G	Cold	
37	Containment Purge Supply Duct - VENT	-13	A	But-terfly	Air	Yes	Closed	Open[b]	Closed	FC	T*	-	No	G	Cold	*Also isolated on high radiation signal.
			B	But-terfly	Air	Yes	Closed	Open[b]	Closed	FC	T*	-	No	G	Cold	
38	Containment Purge Exhaust Duct - VENT	-13	A	But-terfly	Air	Yes	Closed	Open[b]	Closed	FC	T*	-	No	G	Cold	*Also isolated on high radiation signal.
			B	But-terfly	Air	Yes	Closed	Open[b]	Closed	FC	T*	-	No	G	Cold	

[b] If cold and not refueling with Hi Humidity

TABLE 6.2.4-1 (Cont'd)

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PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
39	Plant Air Supply	-14	A	Dia.	Man.	No	Closed	Closed	Closed	As is	No	A	No	G	Cold	Locked closed during power operation - under
	Header		B	Dia.	Man.	No	Closed	Closed	Closed	As is	No	A	No	G	Cold	administrative control.
	(Sec. Sys.)		C	Dia.	Man	No	Closed	Closed	Closed	As is	No	-	No	G	Cold	Can be closed to prevent loss of seal water if there are indications that valve B is not closed tightly.
40	Post Accident Containment	-12	A	Dia.	Air	No	Closed	Closed	Closed	FC	No	-	Yes	G	Cold	
	Venting System		B	Dia.	Air	No	Closed	Closed	Closed	FC	No	-	Yes	G	Cold	
41	Containment	-14	A	But-	Air	Yes	Closed	Closed	Closed	FC	T*	-	No	G	Cold	*Also closed on high
	Pressure		B	terfly	Air	Yes	Closed	Closed	Closed	FC	T*	-	No	G	Cold	radiation signal.
	Relief		C	Dia.	Air	No	Closed	Closed	Closed	FC	No	-	Yes	G	Cold	
			D	Dia.	Air	No	Closed	Closed	Closed	FC	No	-	Yes	G	Cold	
42	Containment	-14	A	But-	Air	Yes	Closed	Closed	Closed	FC	T*	-	No	G	Cold	*Also closed on high
	Vacuum Relief		B	terfly	Air	Yes	Closed	Closed	Closed	FC	T*	-	No	G	Cold	radiation signal.
43	Safety In-	-15	A	Globe	Mot.	Yes	Closed	Closed	Open	As is	No	-	Yes	W	Hot/Cold	
	jection Line		B	Globe	Man.	No	Open	Open	Open	As is	No	-	Yes	W	Hot/Cold	
	- SIS		C	DDV	Mot.	Yes	Closed	Closed	Closed	As is	No	M	Yes	W	Cold	7
			D	Gate	Mot.	Yes	Open	Open	Open*	As is	No	-	Yes	W	Cold	*Closed by operator when
			E	Gate	Mot.	Yes	Open	Open	Open*	As is	No	-	Yes	W	Cold	post-accident recirculation is initiated.

TABLE 6.2.4-1 (Cont'd)

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PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
44	Containment	-16	F	DDV	Man.	No	Open	Open	Open	As is	No	M	Yes	W	Cold	
45	Spray Head-		A	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	W	Cold	*Closed by operator
	ers - SIS		B	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	W	Cold	post-accident recircu-
			C	Check	-	No	Closed	Closed	Open	As is	No	-	Yes	W	Cold	lation is initiated.
			D	Gate	Mot.	Yes	Open	Open	Open*	As is	No	-	Yes	W	Cold	
			E	Gate	Mot.	Yes	Open	Open	Open*	As is	No	-	Yes	W	Cold	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Closing valves D and E makes this a closed system.
46	Containment	-4	A	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	Hot	*Opened by operator for
47	Sump Recir-		B	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	Hot	post accident recircu-
	culation		C.S.	-	-	-	-	-	-	-	-	-	-	-	-	lation RHR System.
	Lines - SIS															
48	Safety In-	-15	A	Globe	Man.	No	L.C.	Closed	Closed	As is	No	A	No	W	Cold	
	jection Test		B	Globe	Man.	No	L.C.	Closed	Closed	As is	No	A	No	W	Cold	
49	Ventil. Sys-	-16	C.S.	-	-	-	-	-	-	-	-	-	-	-	-	Containment cooler and its cooling water piping are missile protected and a closed system inside containment.
50	tem Cooling															
51	Water In-															
52	VENT															
			A	Butter-fly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	Cold	

TABLE 6.2.4-1 (Cont'd)

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PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
65	Accumulator	-17	A	Globe	Air	Yes	Open	Closed	Closed	FC	T	-	No	G	Cold	
	Nitrogen		B	Check	--	No	Open	Open	Closed	As is	No	-	No	G	Cold	
	Supply															
66	Containment	-18	A	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	G	Cold	
	Test Channel		B	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	G	Cold	
	Line															
67	Containment	-19	A	Globe	Man.	No	Closed	Closed*	Closed	As is	No	-	No	G	Cold	*Open for CV pressure test
	Test Control-led Leakage		B	Globe	Man.	No	Closed	Closed*	Closed	As is	No	-	No	G	Cold	
68	Containment															
69	Pressure		A	Globe	Man.	No	Open	Open	Open	As is	No	-	No	G	Cold	
70	Sensing	-18	B	Globe	Man.	No	Open	Open	Open	As is	No	-	N	G	Cold	
	Lines															
71	Penetration	-19	A	Globe	Man.	No	Open	Open	Open	As is	No	-	No	G	Cold	Maintains PPS to pene-trations inside CV
	Pressure Air		B	Globe	Man.	No	Closed	Closed	Closed	As is	No	-	No	G	Cold	
	Supply		C	Globe	Man.	No	Open	Open	Open	As is	No	-	No	G	Cold	

TABLE 6.2.4-1 (Cont'd)

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PPS - Penetration Pressurization System

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	TEMP	NOTES
72	Dead-weight Tester Line	-17	A	Globe	Man.	No	Closed	Closed	Closed	As Is	No	-	No	W	Cold	
			B	Globe	Man.	No	Closed	Closed	Closed	As Is	No	-	No	W	Hot	
A	Containment Fire Water	-20	A	Gate	Mot.	Yes	Open	Open	Closed	As Is	T	-	No	W	Cold	Penetration numbers not assigned
			B	Globe	Man.	No	Closed	Closed	Closed	As Is	No	-	No	W	Cold	
			C	Gate	Mot.	Yes	Open	Open	Closed	As Is	T	-	No	W	Cold	
B	Containment Fire Water	-20	A	Gate	Mot.	Yes	Open	Open	Closed	As Is	T	-	No	W	Cold	
			B	Globe	Man.	No	Closed	Closed	Closed	As Is	No	-	No	W	Cold	
			C	Gate	Mot.	Yes	Open	Open	Closed	As Is	T	-	No	W	Cold	

TABLE 6.2.4-2

AUTOMATIC ISOLATION VALVE SIZES

VALVE SIZE	PENETRATION NO.
3/8 in.	1, 29, 30, 31, 60
3/4 in.	5, 67
1 in.	4, 35, 36
2 in.	13, 14, 15, 23, 33, 62, 63, 64
3 in.	3, 20, 22, 28, 43
6 in.	18, 19, 41, 42
10 in.	17
26	7, 8, 9
42	37, 38

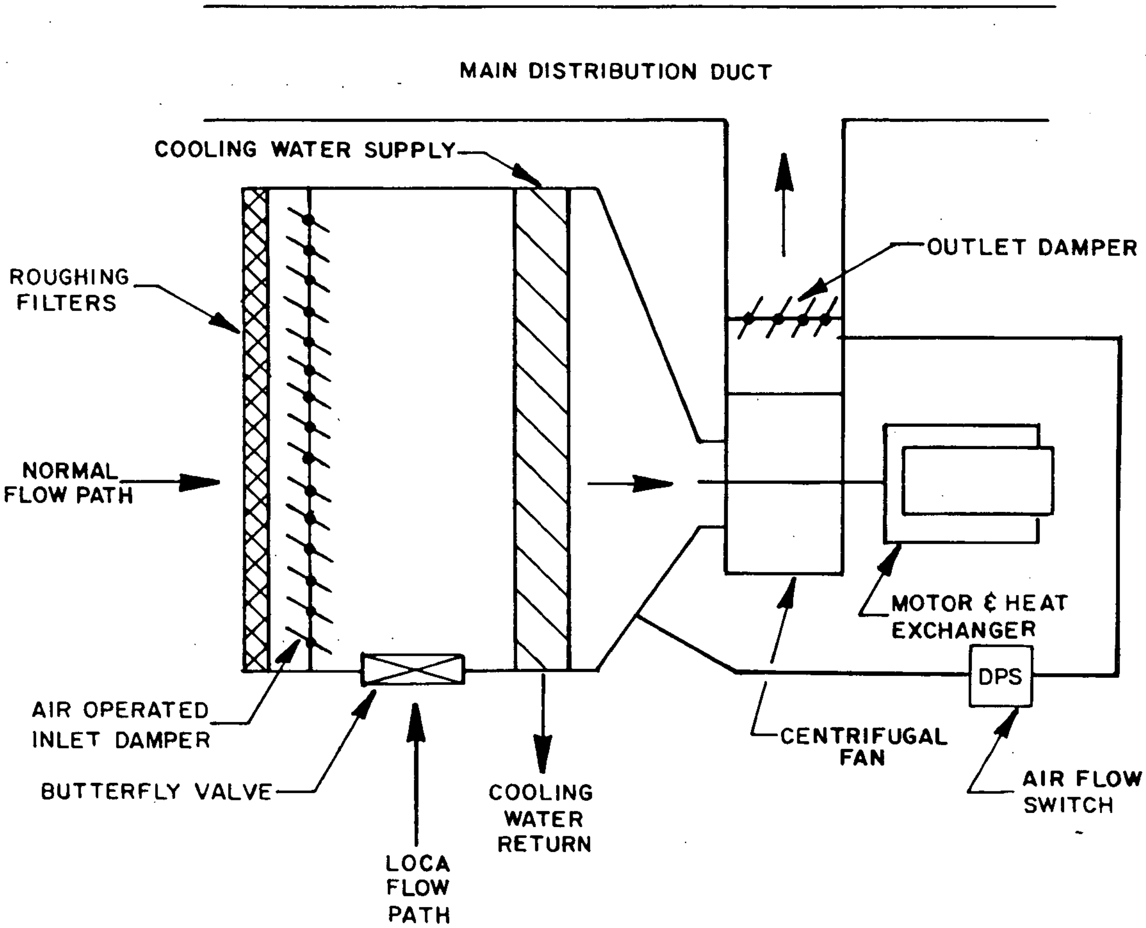
REFERENCES: SECTION 6.2 (Cont'd)

6.2.5-1 K. G. Davenport, et al. "Control of Hydrogen Concentration Following a Loss-of-Coolant Accident by Containment Venting for the H. B. Robinson Plant," WCAP-7372, REV. 1, August 1970.

6.2.6-1 Letter, No Serial Number, dated July 23, 1970, w/enclosure: CP&L Report dated July 23, 1970, H. B. Robinson Steam Electric Plant, Unit 2, entitled, "Pre-operational Integrated Leakage Rate and Sensitive Leakage Rate Test of the Reactor Containment Building."

6.2.6-2 Letter, No Serial Number, dated July 28, 1970, from CP&L to USAEC w/enclosure: Addendum I to H. B. Robinson Unit 2 Report dated July 23, 1970, entitled "Pre-operational Integrated Leakage Rate and Sensitive Leakage Rate Test of Reactor Containment Building."

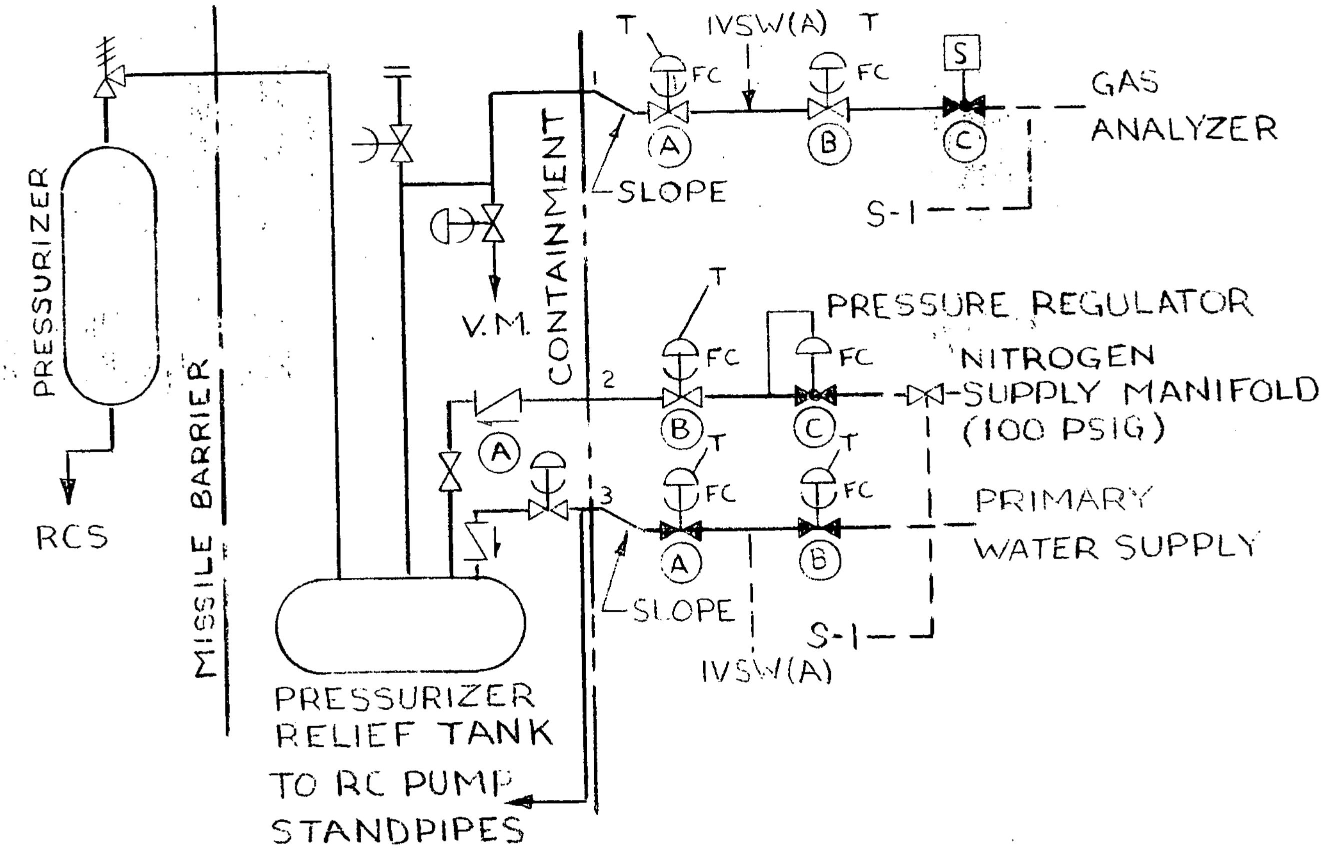
6.2.6-3 Letter, GD-78-1925, July 12, 1978, from CP&L to NRC w/enclosure: GAI Report No. 1976 on Test Performed February, 1978, entitled "Integrated Leak Rate Test of the Reactor Containment Building," H. B. Robinson Steam Electric Plant, Unit 2, issued May 15, 1978.



AMENDMENT 3

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT AIR RECIRCULATION COOLING SYSTEM	FIGURE 6.2.2 - 2

PENE. NO. 1 - PRESSURIZER RELIEF GAS ANALYZER LINE
PENE. NO. 2 - PRESSURIZER RELIEF NITROGEN SUPPLY
PENE. NO. 3 - PRESSURIZER RELIEF TANK MAKEUP



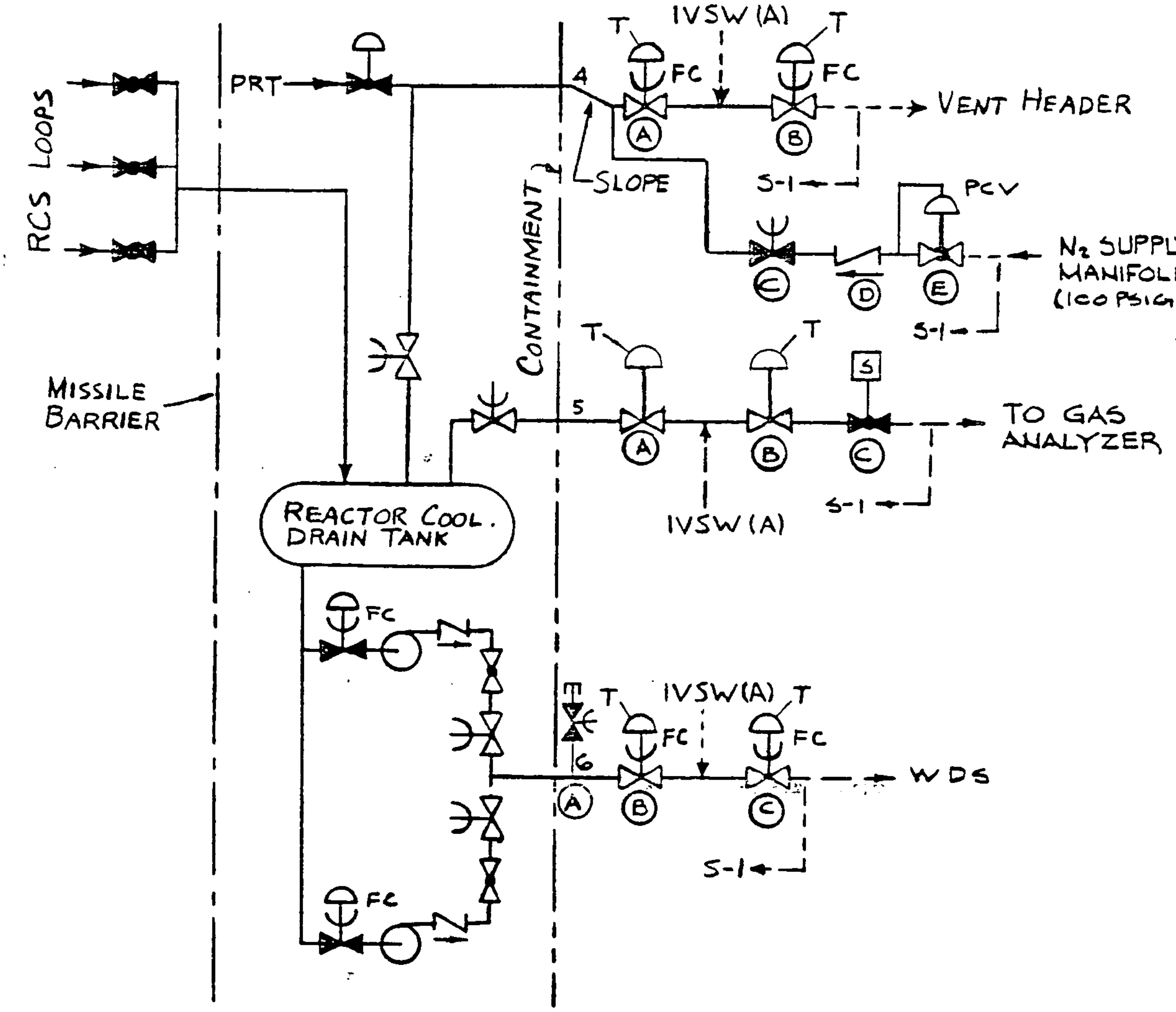
1. ALTHOUGH THE PRESSURIZER RELIEF TANK IS MISSILE PROTECTED, THESE PENETRATING LINES CAN BECOME EXPOSED TO CONTAINMENT ATMOSPHERE IF THE PRESSURIZER SAFETY VALVE DISCHARGE HEADER IS BREACHED BY THE ACCIDENT.
2. VALVE C IN THE LINE TO THE GAS ANALYZER IS OPENED INTERMITTENTLY TO WITHDRAW A GAS SAMPLE. THIS OPERATION IS AUTOMATICALLY CONTROLLED BY THE GAS ANALYZER.

FOR LIST OF SYMBOLS SEE FIG. 6.2.4-21

AMENDMENT 3

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PRESSURIZER RELIEF TANK	FIGURE 6.2.4 - 1

PENE. NO. 4 - PRIMARY SYSTEM VENT HEADER (& N_2 SUPPLY LINE)
PENE. NO. 5 - REACTOR COOLANT DRAIN TANK GAS ANALYZER LINE
PENE. NO. 6 - REACTOR COOLANT DRAIN TANK PUMP DISCHARGE LINE



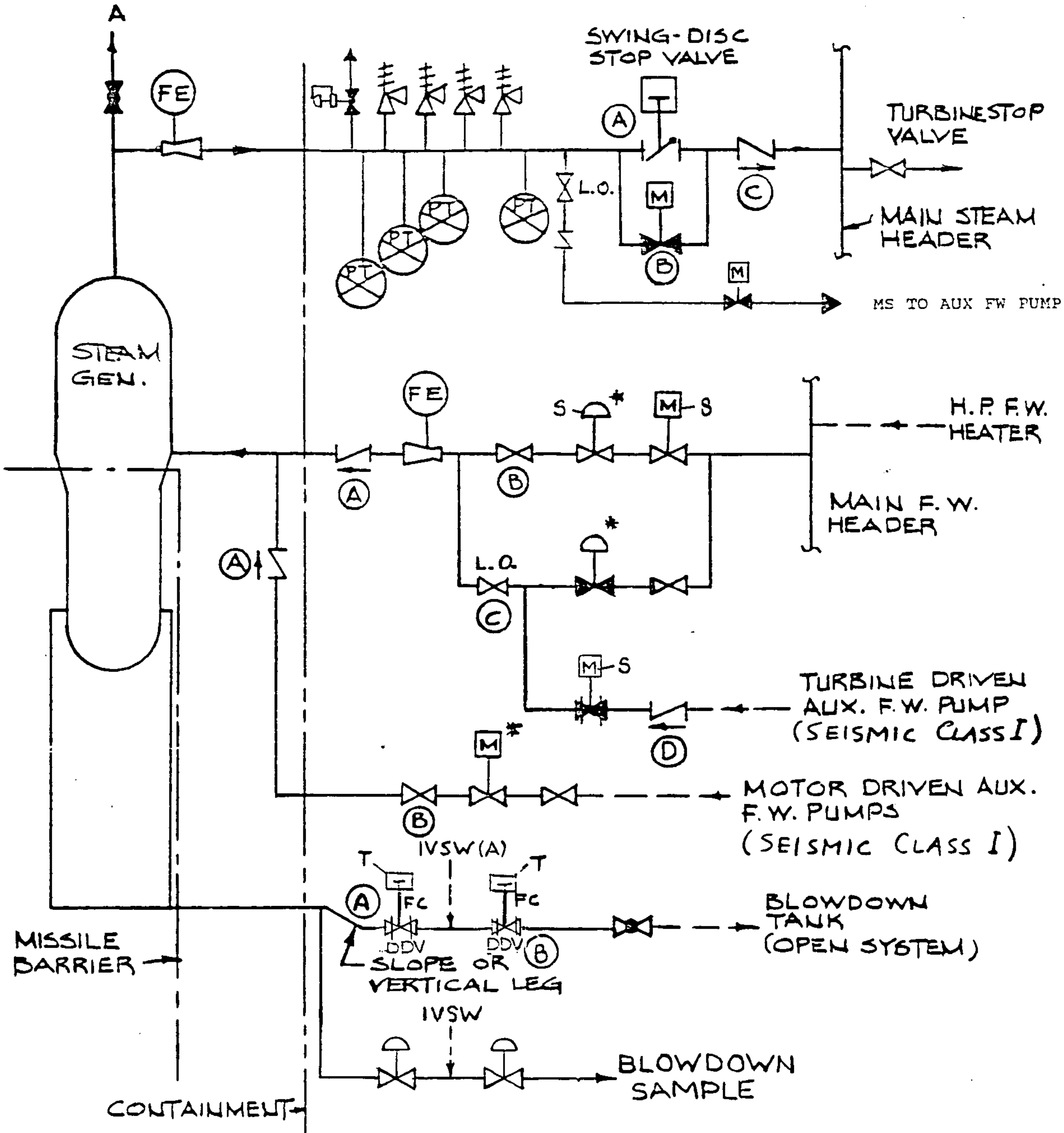
WDS – WASTE DISPOSAL SYSTEM

1. VALVE Ⓒ IN THE LINE TO THE GAS ANALYZER IS OPENED INTERMITTENTLY TO WITHDRAW A GAS SAMPLE. THIS OPERATION IS AUTOMATICALLY CONTROLLED BY THE GAS ANALYZER.

AMENDMENT NO. 6

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES REACTOR COOLANT DRAIN TANK	FIGURE 6.2.4 - 2

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

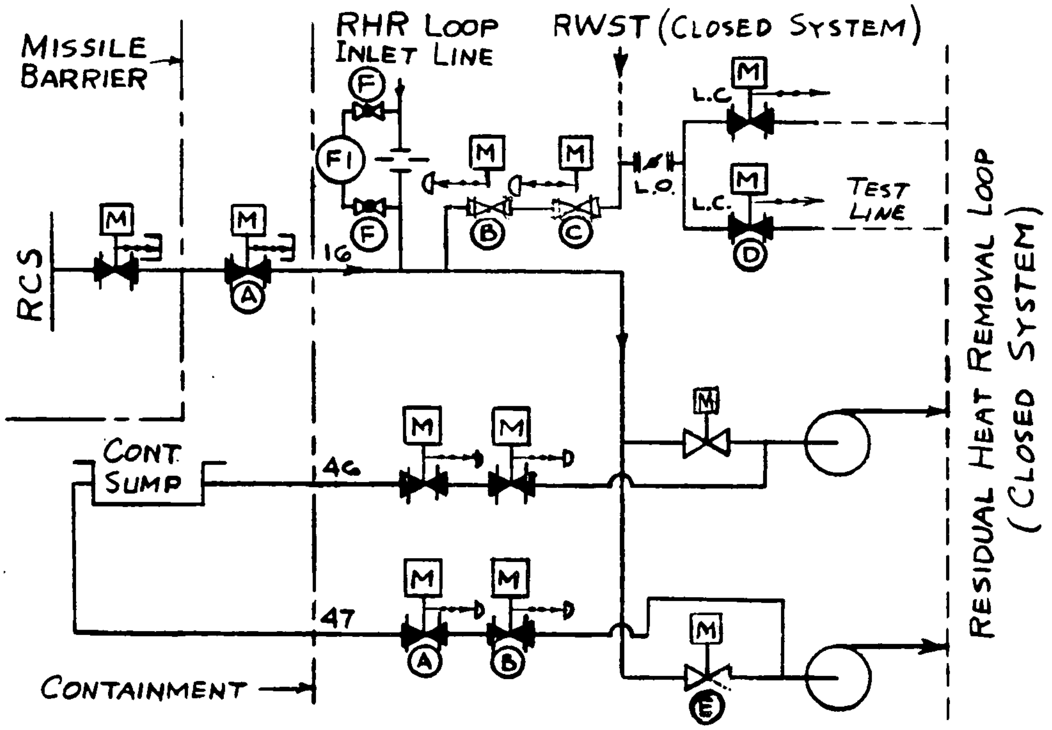


1. * – FEEDWATER REGULATORS

AMENDMENT NO. 6

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

PENE. NO. 16 - RESIDUAL HEAT REMOVAL LOOP OUT
PENE. NO. 46, 47 - CONTAINMENT SUMP RECIRCULATION LINES

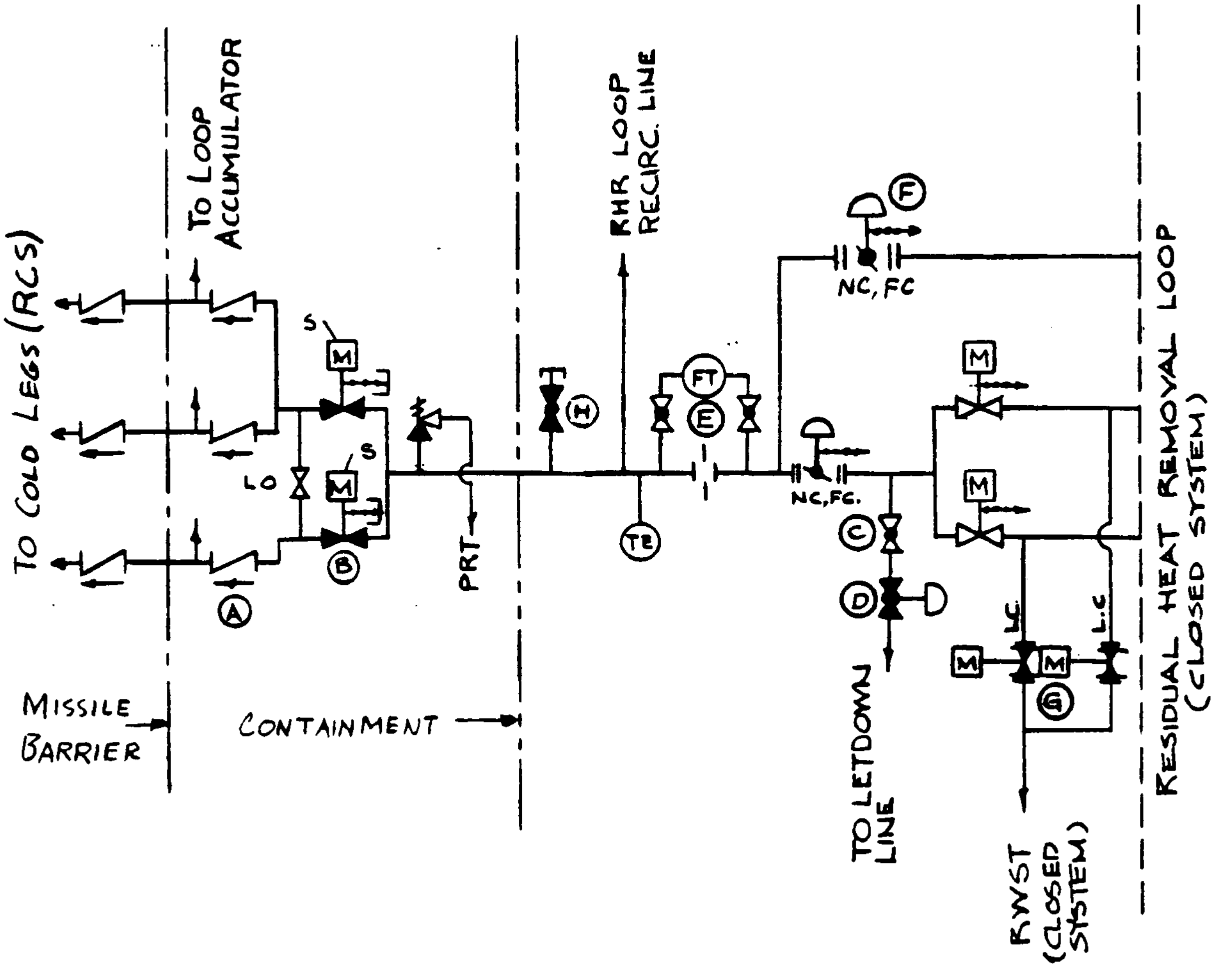


RWST - REFUELING WATER STORAGE TANK
RHR - RESIDUAL HEAT REMOVAL
ENTIRE SYSTEM IS SEISMIC CLASS I

AMENDMENT NO. 10

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES RHR LOOP (OUT) AND SUMP RECIRCULATION	FIGURE 6.2.4 - 4

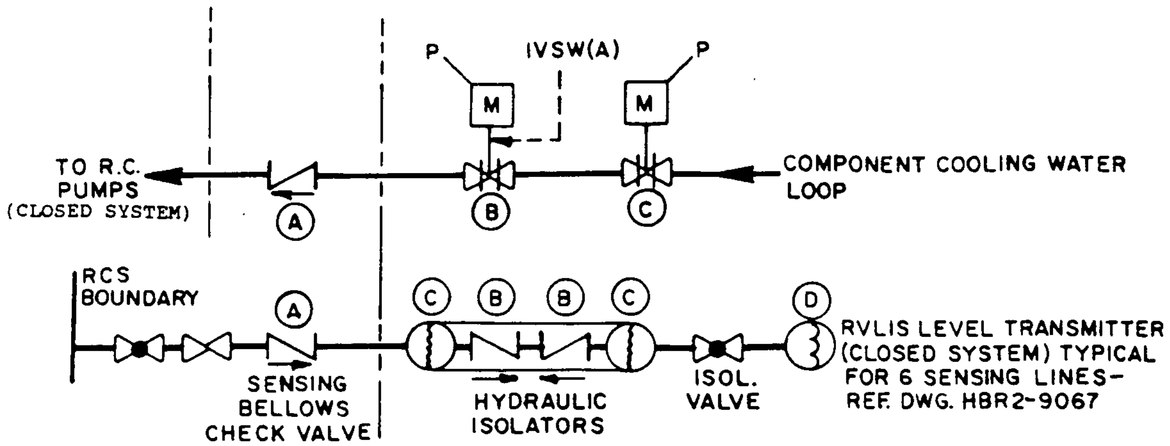
PENE. NO. 17 - RESIDUAL HEAT REMOVAL LOOP IN



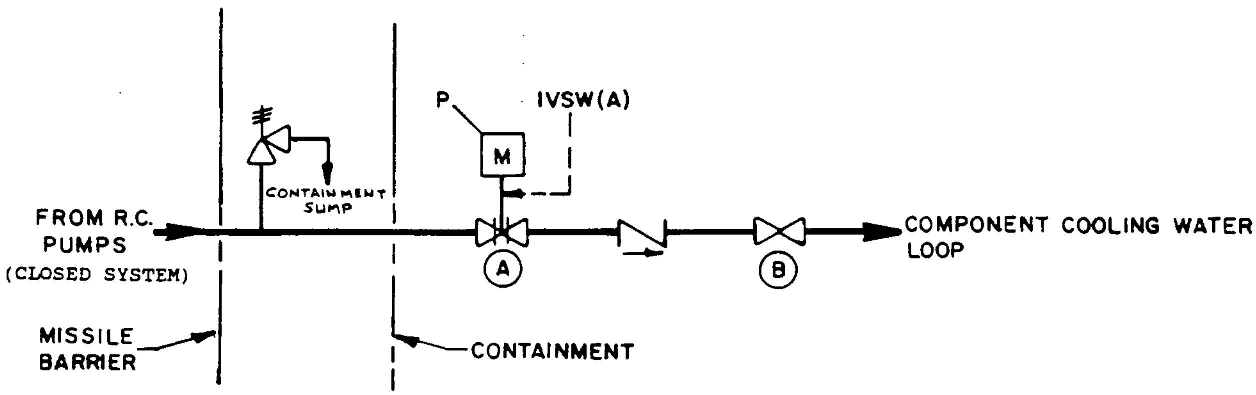
1. VALVE B1 IS IN A LINE WHICH CONNECTS TO THE LETDOWN LINE INSIDE CONTAINMENT.
2. ENTIRE SYSTEM IS SEISMIC CLASS I

AMENDMENT NO. 9

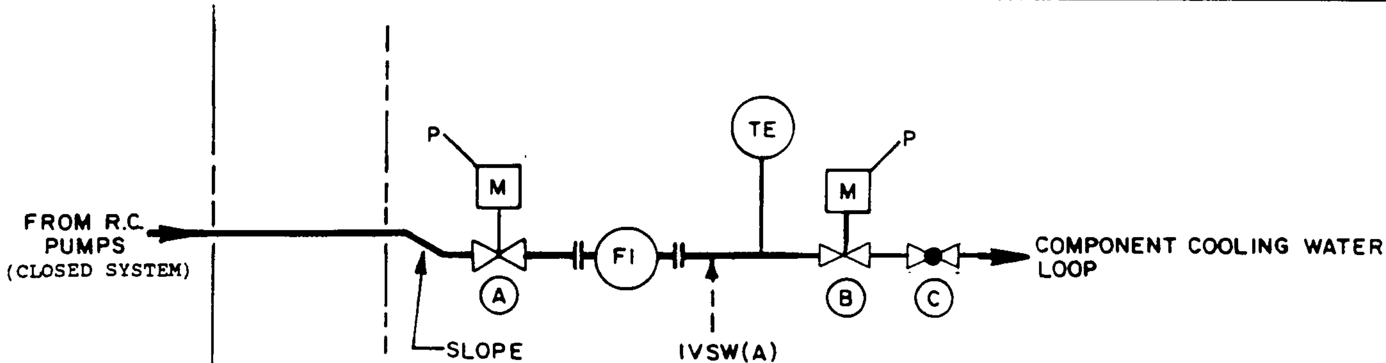
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES RHR LOOP (IN)	FIGURE 6.2.4 - 5



PENE. NO.18-REACTOR COOLANT PUMP COOLING WATER IN- & RVLIS SENSING LINES



PENE. NO. 19-REACTOR COOLANT PUMP COOLING WATER OUT (6")



PENE. NO. 20-REACTOR COOLANT PUMP COOLING WATER OUT (3")

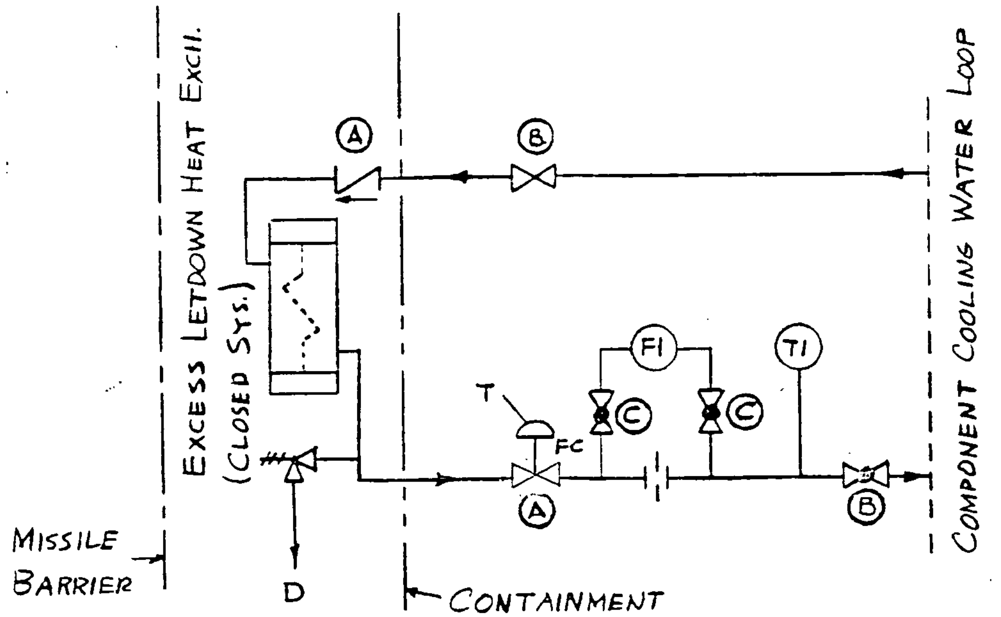
COMPONENT COOLING WATER LOOP IS SEISMIC CLASS I.

AMENDMENT NO. 9

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES R. C. PUMP COOLING WATER AND RVLIS SENSING LINES	FIGURE 6.2.4 - 6

PENE. NO. 21 - EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN

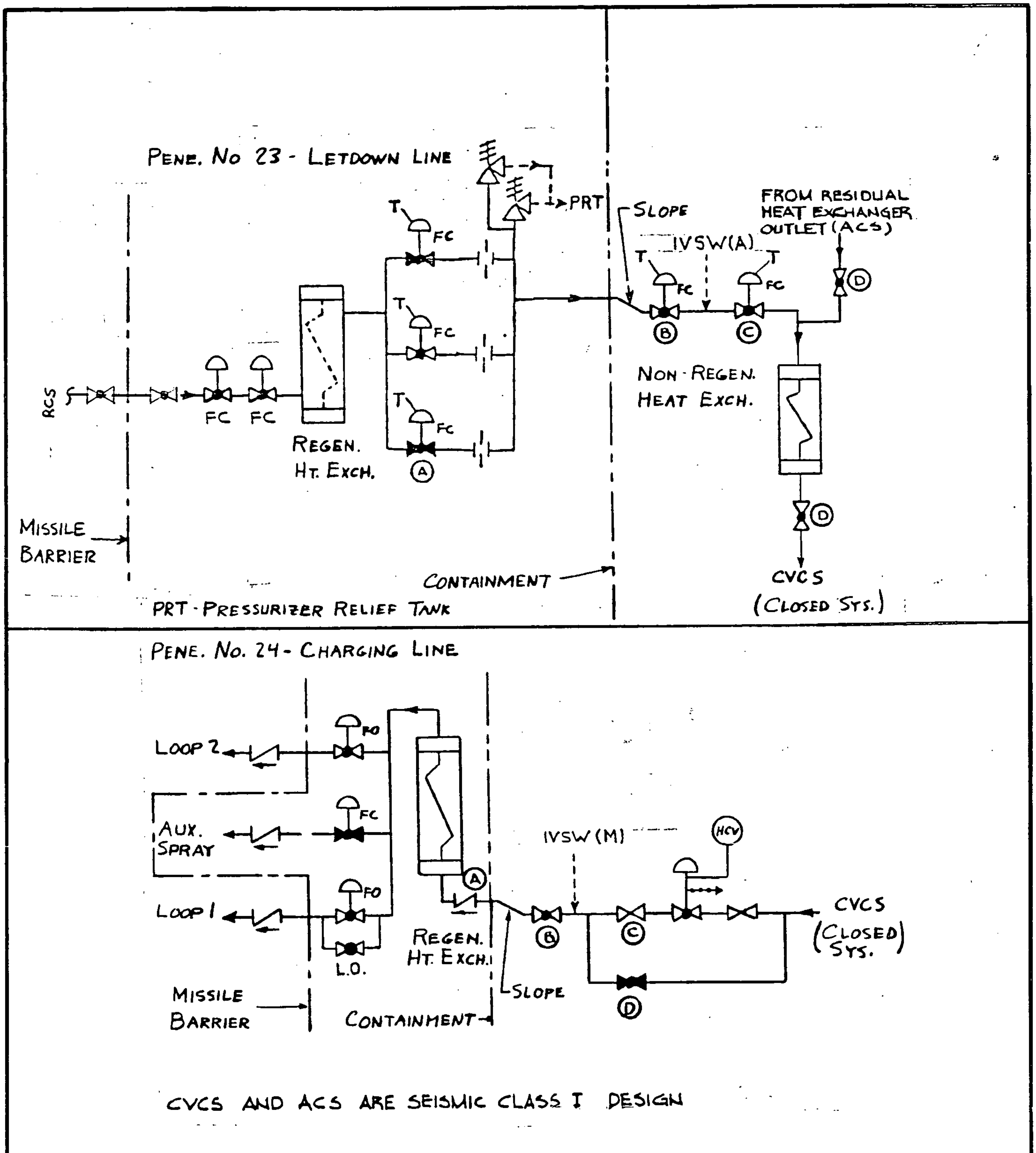
PENE. NO. 22 - EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT



COMPONENT COOLING WATER LOOP IS SEISMIC CLASS I

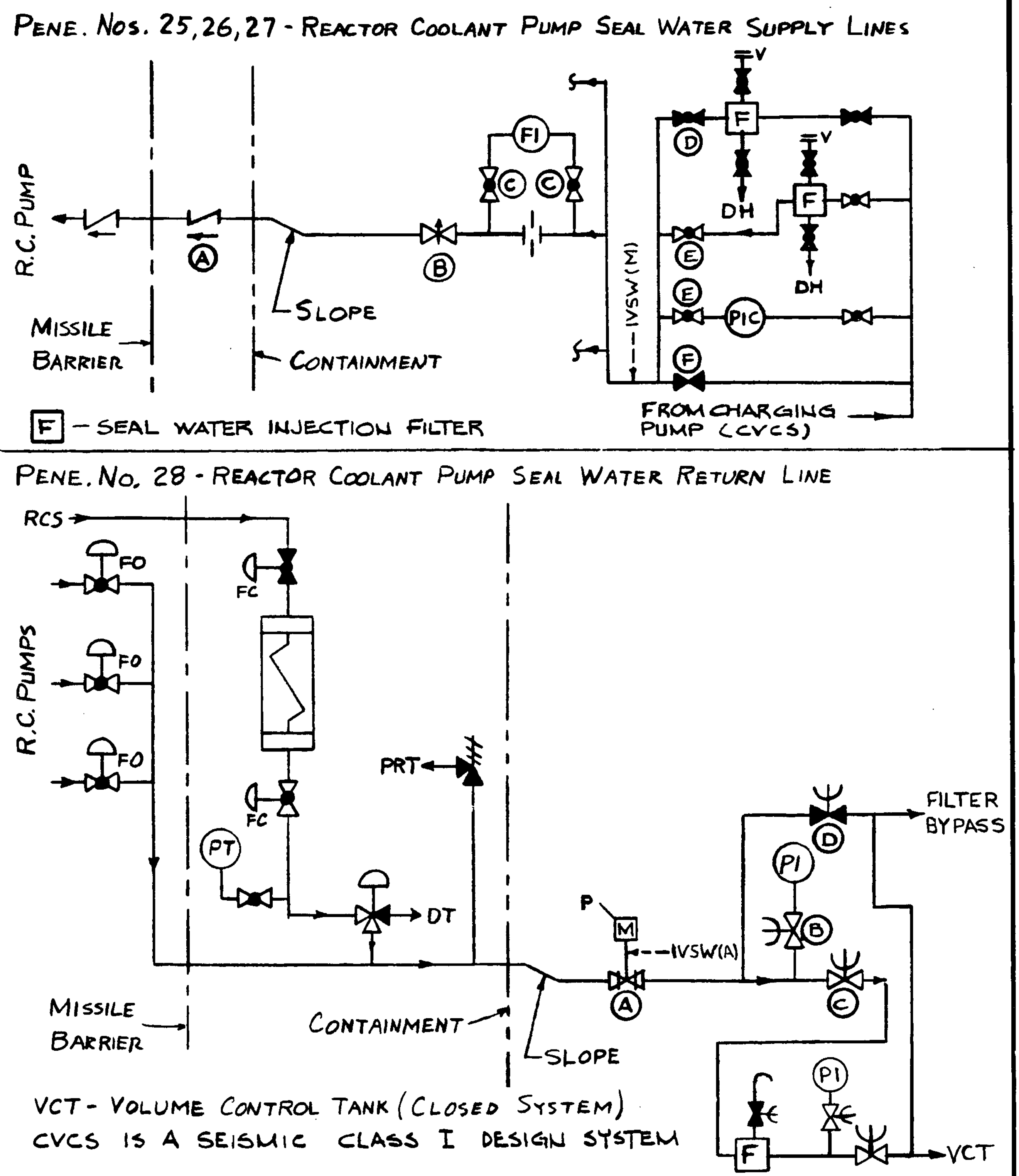
AMENDMENT NO. 6

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES EXCESS LETDOWN COOLING IN AND OUT	FIGURE 6.2.4 - 7

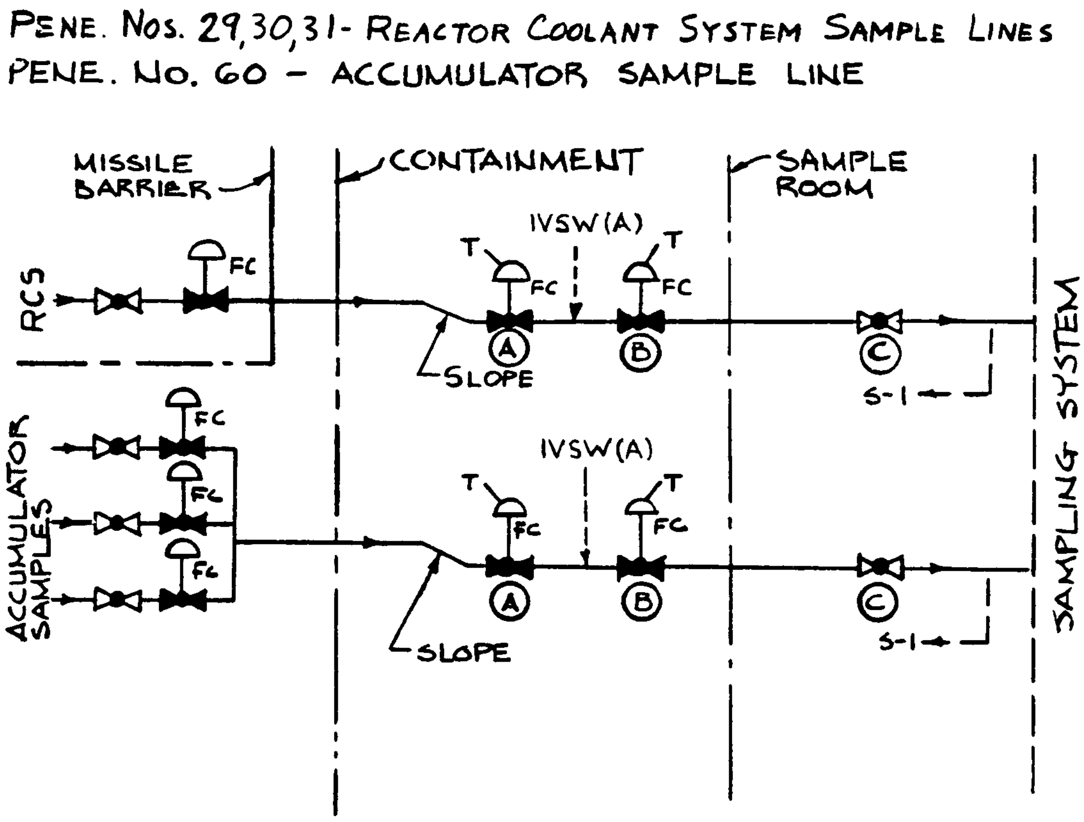


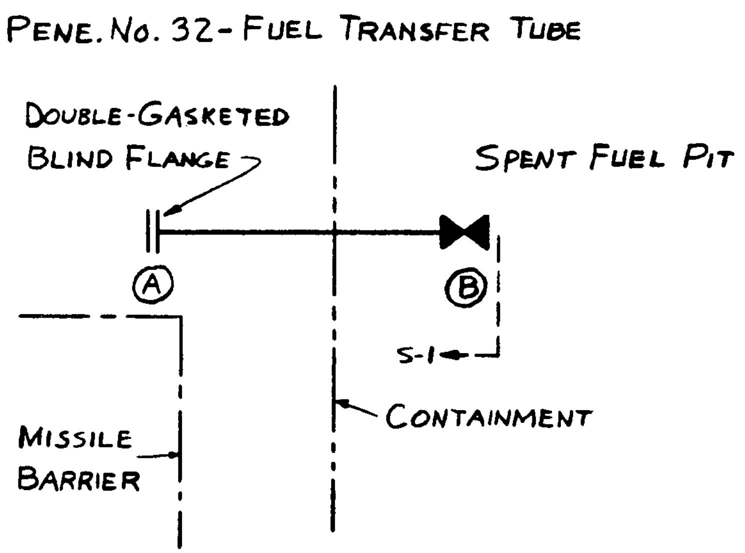
AMENDMENT NO. 6

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES LETDOWN LINE	FIGURE 6.2.4 - 8



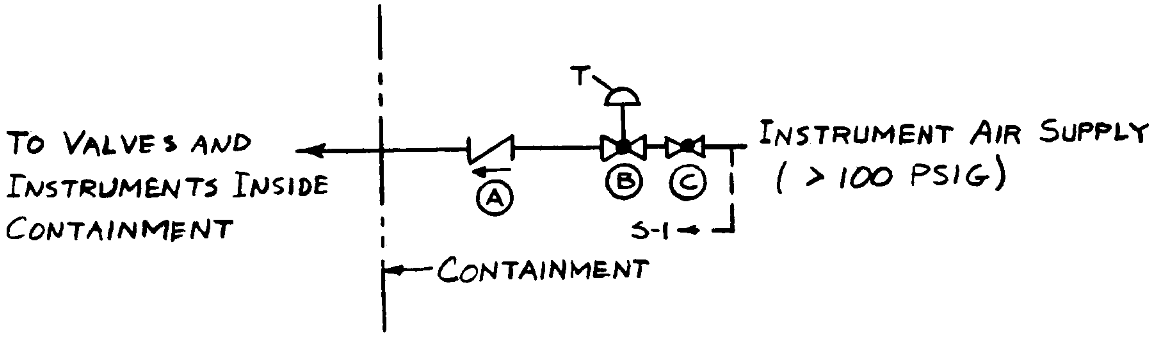
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES R. C. PUMP SEAL WATER	FIGURE 6.2.4 - 9



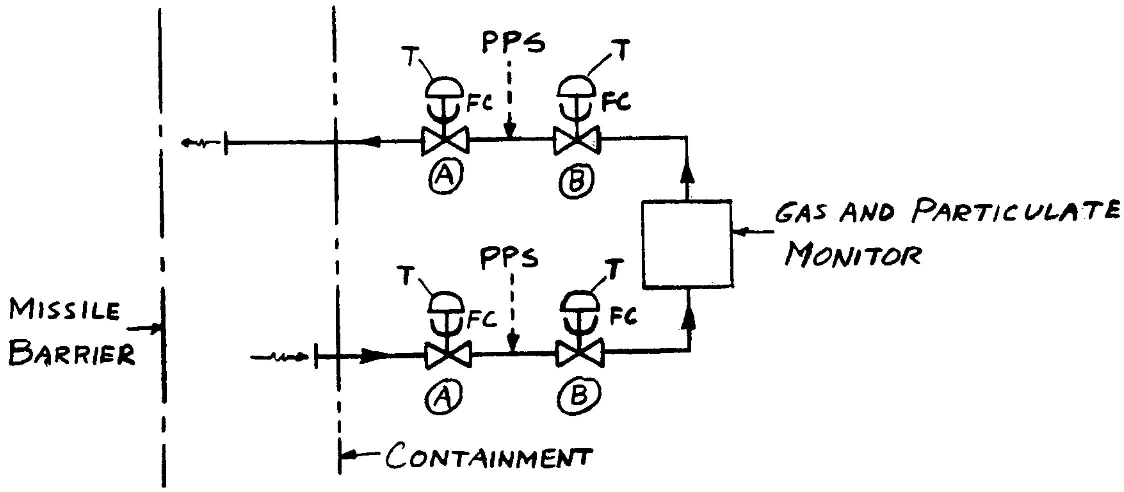


H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES R. C. SYSTEM SAMPLE LINES AND FUEL TRANSFER TUBE	FIGURE 6.2.4 - 10

PENE. NO. 33 - INSTRUMENT AIR HEADER



PENE. NO. 35 - CONTAINMENT AIR SAMPLE IN
PENE. NO. 36 - CONTAINMENT AIR SAMPLE OUT



1. CONTAINMENT ISOLATION SIGNAL APPLIED TO THESE VALVES MUST BE OVERRIDDEN IN ORDER TO USE THE MONITOR AFTER AN ACCIDENT.
2. PPS – PENETRATION PRESSURIZATION SYSTEM.
3. ENTIRE SYSTEM IS SEISMIC CLASS I DESIGN.

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES IA HEADER AND CONTAINMENT AIR SAMPLE	FIGURE 6.2.4 - 11

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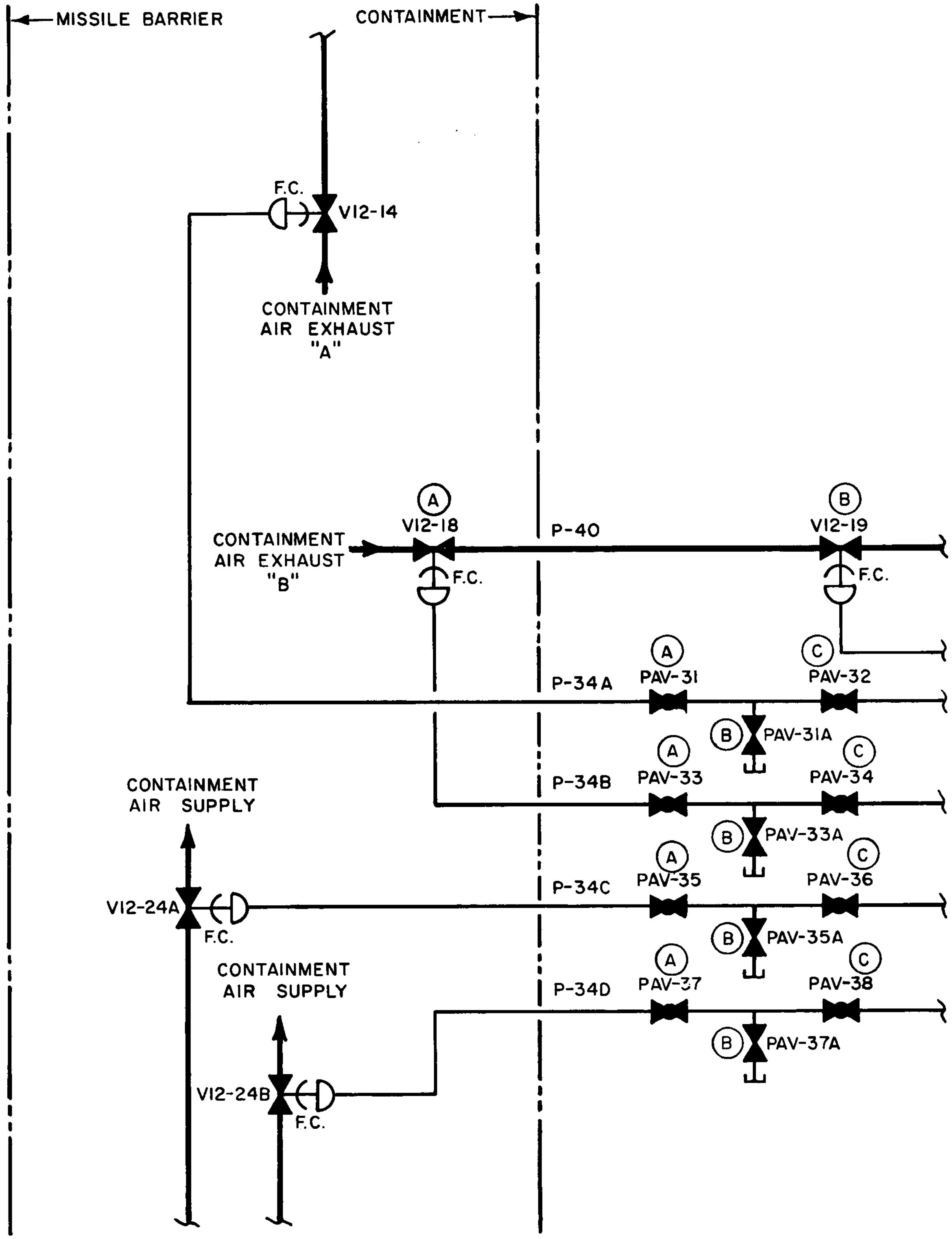
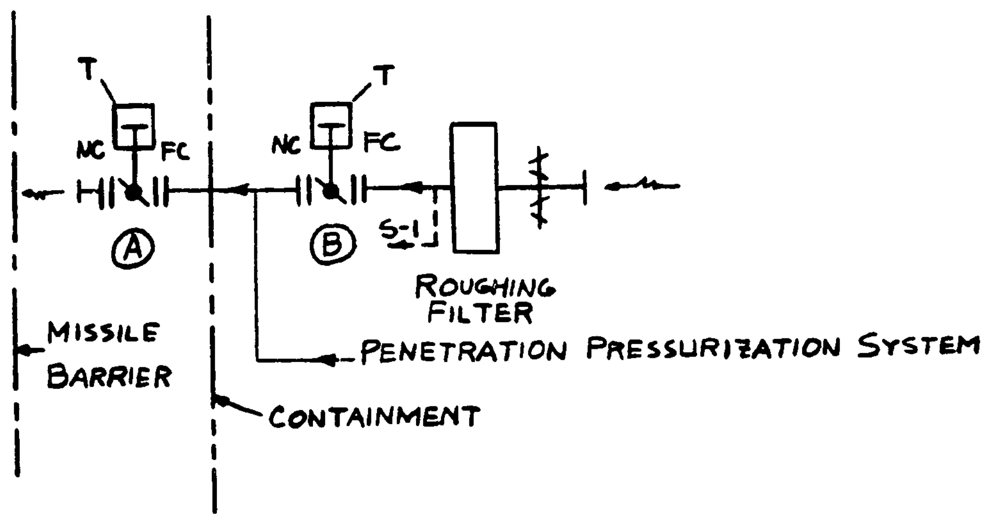


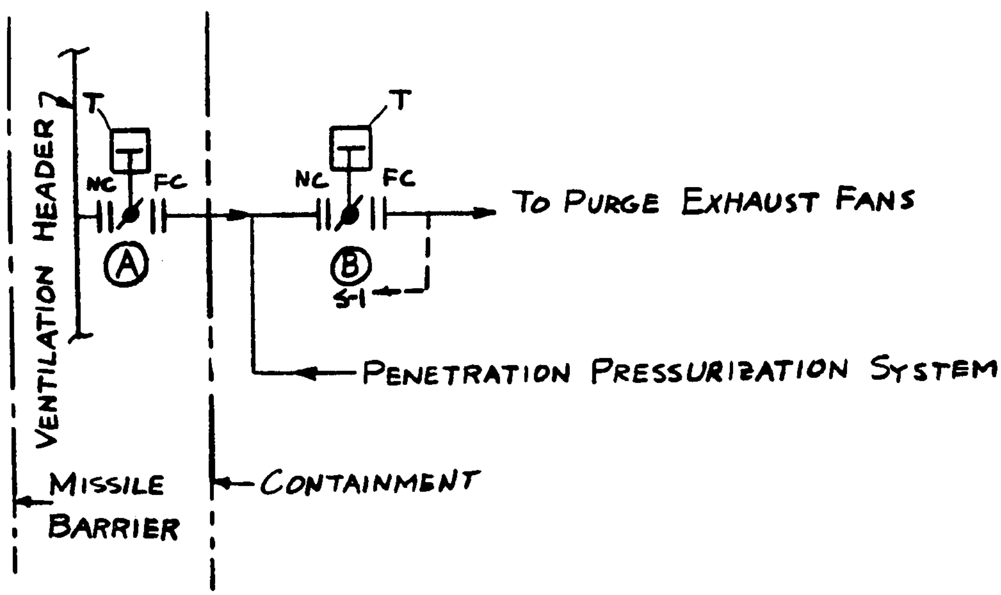
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES POST ACCIDENT VENTING SYSTEM	FIGURE 6.2.4 - 12

PENE. No. 37 - CONTAINMENT PURGE SUPPLY DUCT



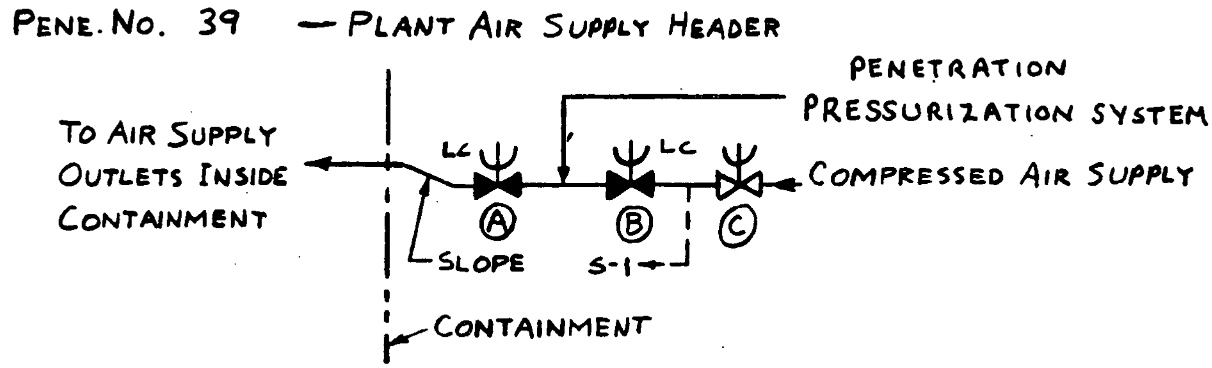
1. THESE VALVES HAVE AIR CYLINDER OPERATORS.

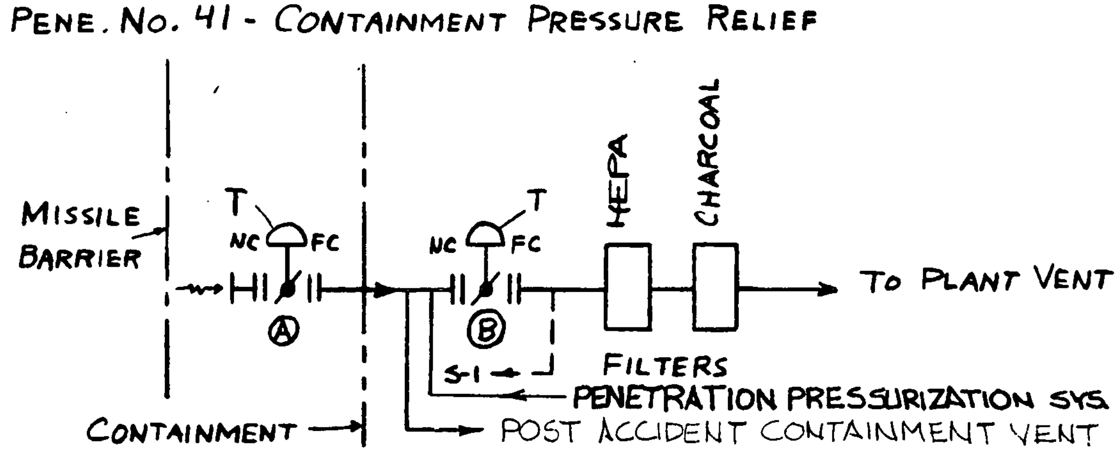
PENE. No. 38 - CONTAINMENT PURGE EXHAUST DUCT

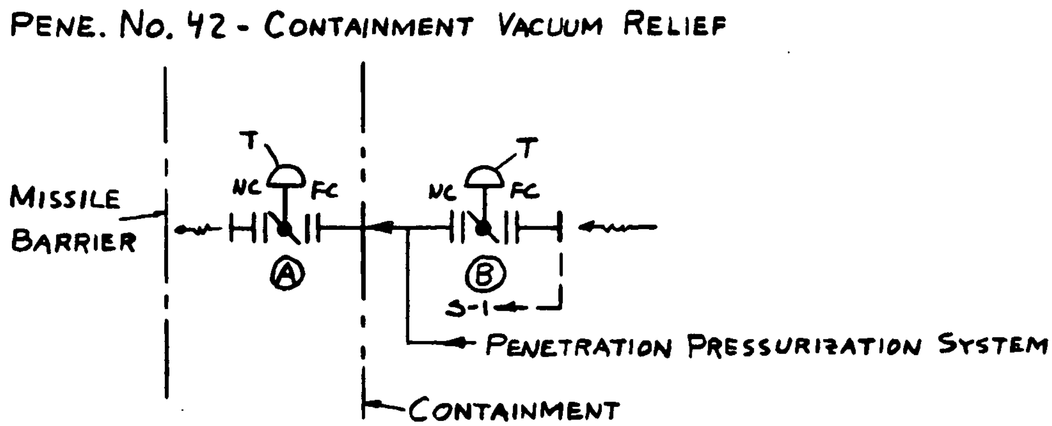


1. THESE VALVES HAVE AIR CYLINDER OPERATORS.

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PURGE SUPPLY AND EXHAUST	FIGURE 6.2.4 - 13



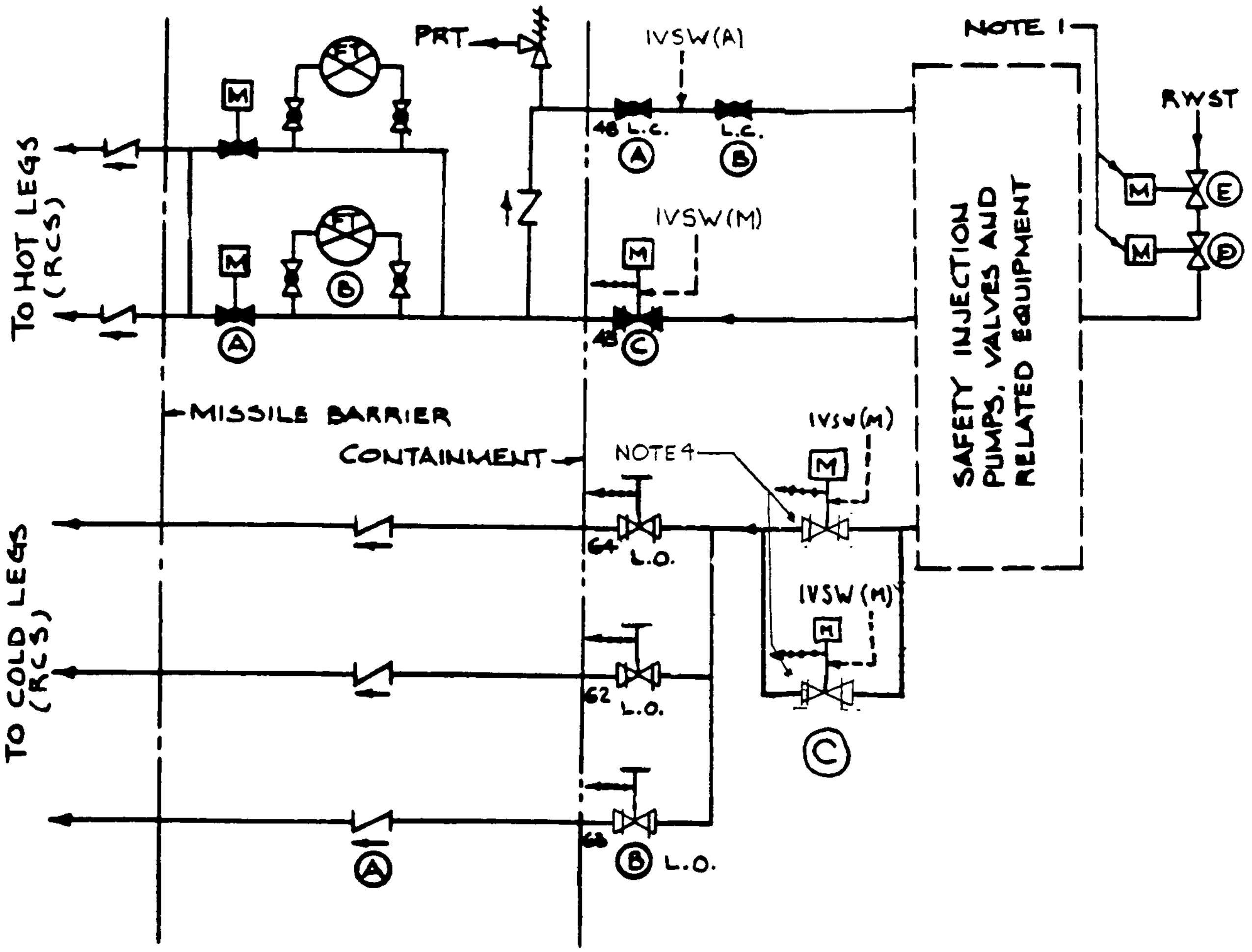




AMENDMENT NO.5

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES SA HEADER, PRESSURE AND VACUUM RELIEF	FIGURE 6.2.4 - 14

PENE. NO. 43 – SAFETY INJECTION LINE
PENE. NO. 48 – SAFETY INJECTION TEST LINE
PENE. NOS. 62, 63, 64 – BORON INJECTION LINES.



NOTE 1 – THESE VALVES CLOSED BY OPERATOR AFTER RWST HAS BEEN EMPTIED BY SAFETY INJECTION

NOTE 2 – SAFETY INJECTION PUMPS AND ASSOCIATED VALVES (EXCEPT CONTAINMENT ISOLATION VALVES SHOWN ABOVE) ARE LOCATED IN ROOM WITH VENTILATION EXHAUST THROUGH CHARCOAL FILTERS FOLLOWING ACCIDENT.

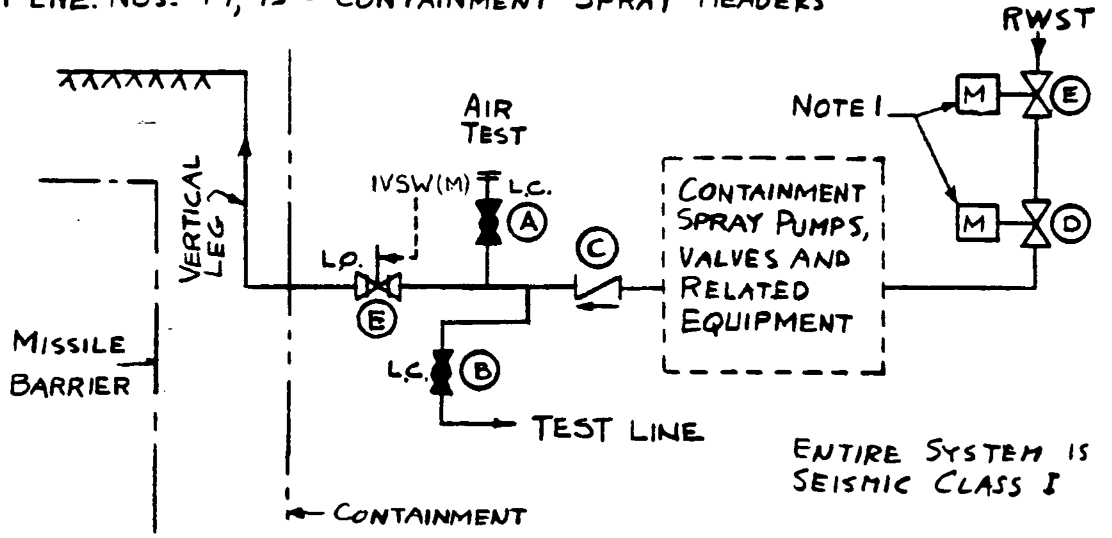
NOTE 3 – ENTIRE SYSTEM IS SEISMIC CLASS I

NOTE 4 - THESE VALVES CAN BE MANUALLY CLOSED BY OPERATOR AFTER RWST HAS BEEN EMPTIED BY SAFETY INJECTION IF MOV IS NOT OPERABLE.

AMENDMENT NO. 11

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES SAFETY INJECTION SYSTEM	FIGURE 6.2.4 - 15

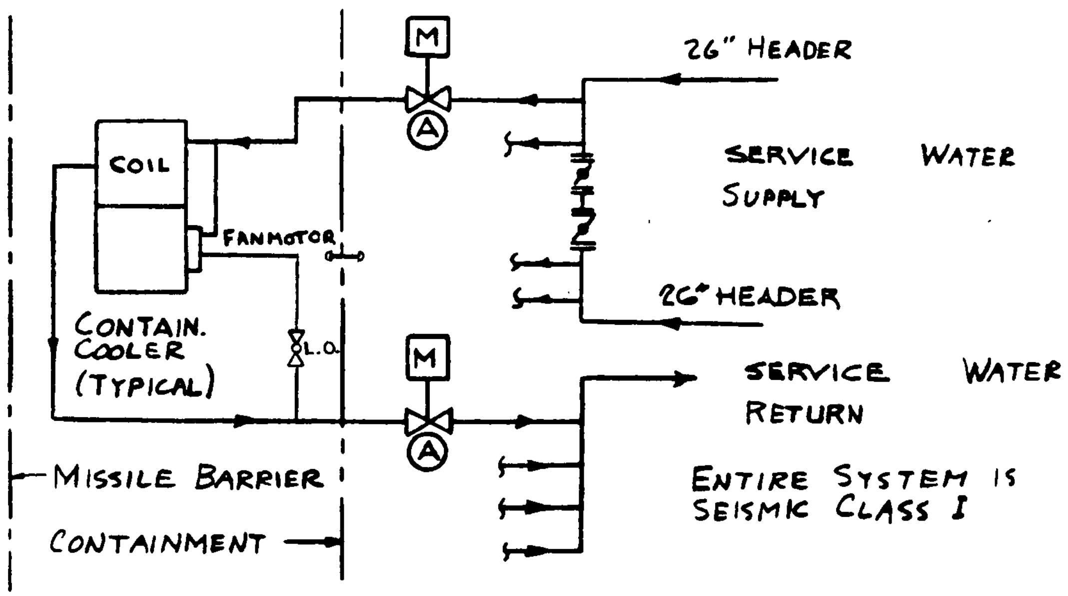
PENE. NOS. 44, 45 - CONTAINMENT SPRAY HEADERS



NOTE 1 - THESE VALVES CLOSED BY OPERATOR AFTER RWST HAS BEEN EMPTIED BY SAFETY INJECTION.

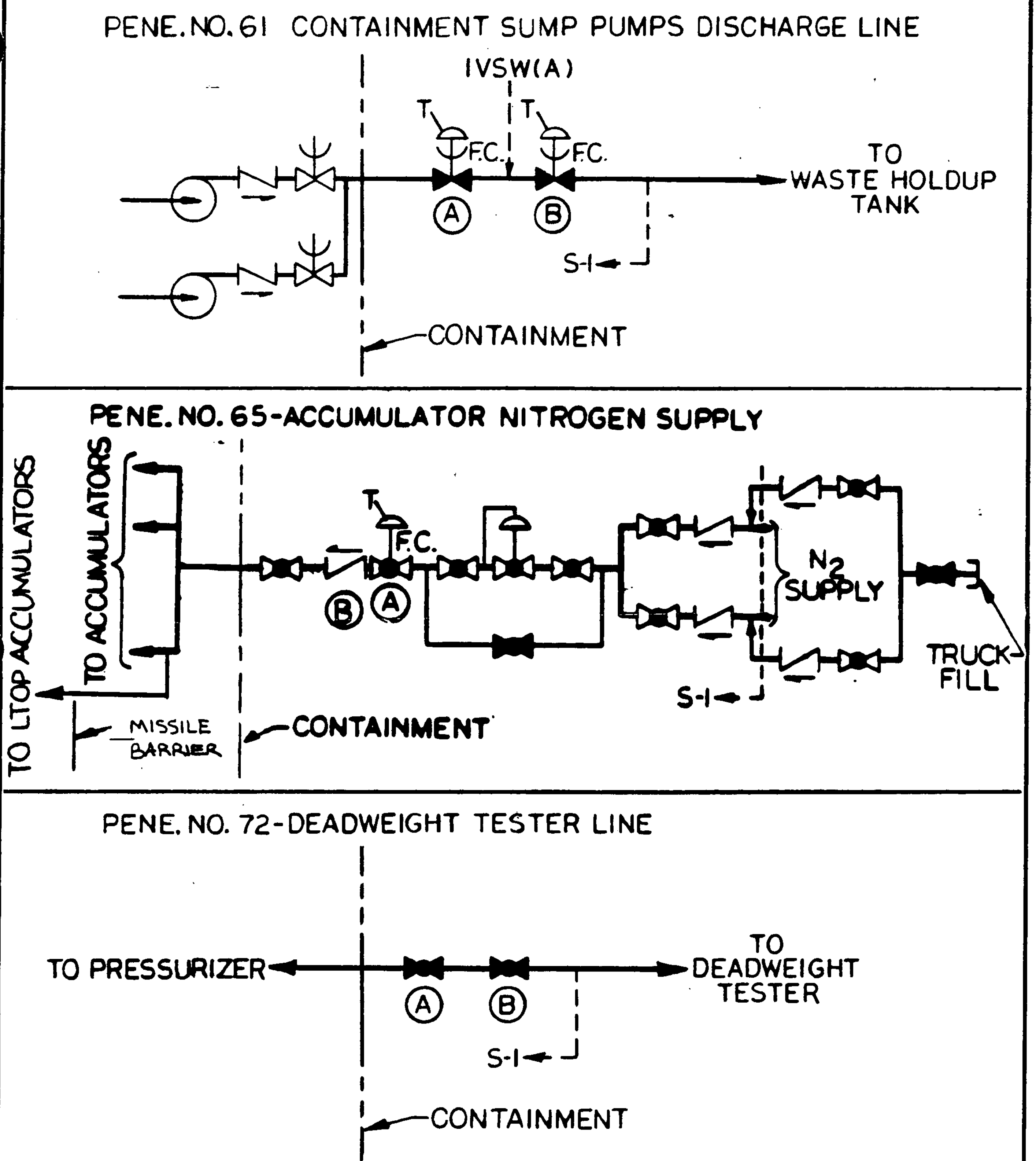
NOTE 2 - CONTAINMENT SPRAY PUMPS AND ALL ASSOCIATED VALVES EXCEPT D, E AND F LOCATED IN ROOM WITH VENTILATION EXHAUST THROUGH CHARCOAL FILTERS FOLLOWING ACCIDENT.

PENE. NOS. 49, 50, 51, 52 - VENTIL. SYSTEM COOLING WATER IN
PENE. NOS. 53, 54, 55, 56 - VENTIL. SYSTEM COOLING WATER OUT - COILS
PENE. NOS. 53A, 54A, 55A, 56A - SPARE



AMENDMENT NO. 7

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES SPRAY HDRS AND VENT. SYSTEM COOLING WATER	FIGURE 6.2.4 - 16

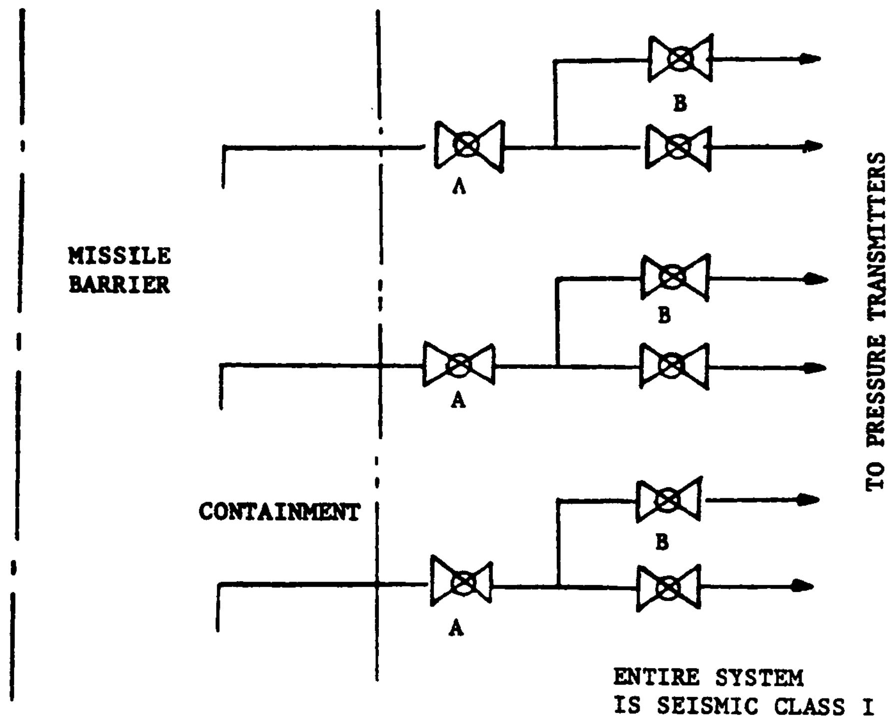


Amendment No. 7

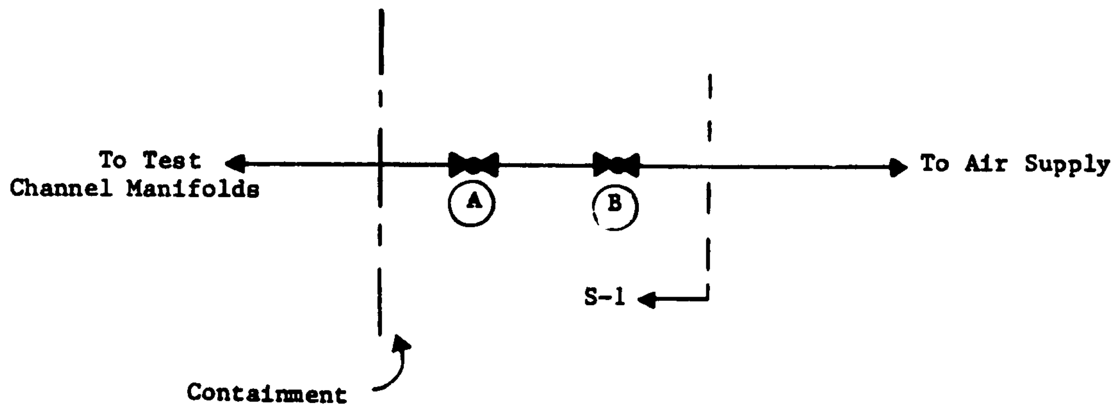
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES SUMP PUMPS DISCHARGE, N_2, AND DEADWEIGHT TESTER	FIGURE 6.2.4 - 17

PENETRATION NUMBERS 68, 69, 70

CONTAINMENT PRESSURE SENSING LINES

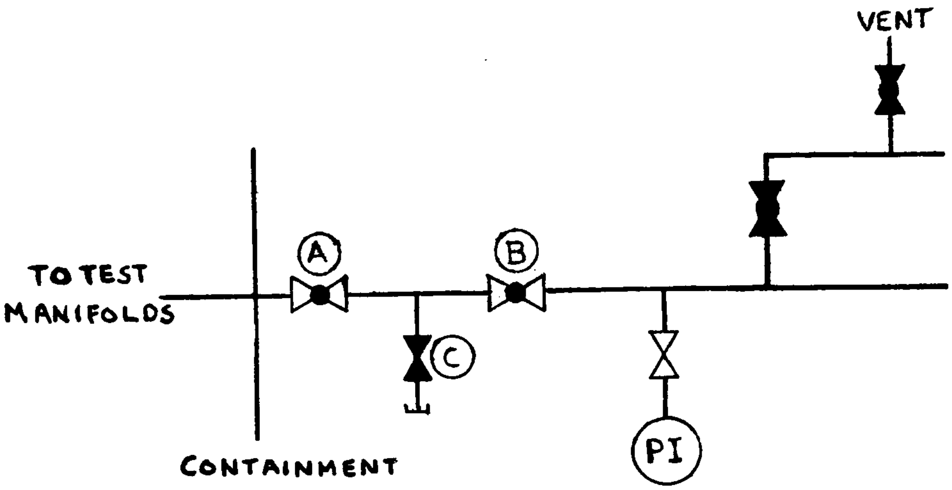


Penetration No. 66 - Containment Test Channel Line



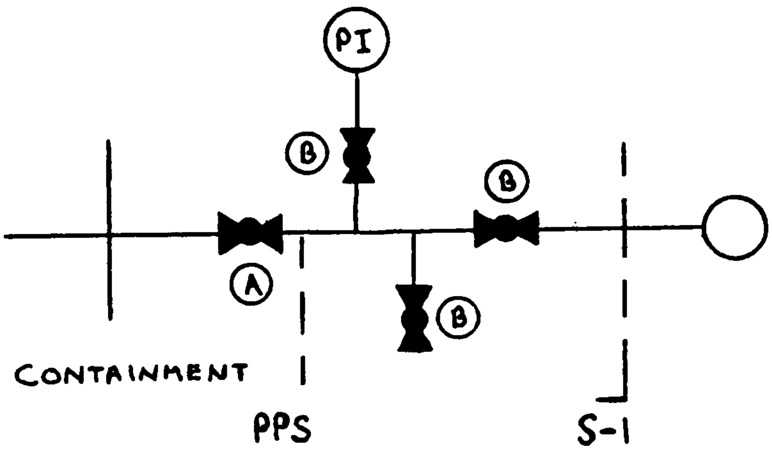
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PRESSURE SENSORS AND TEST CHANNEL LINE	FIGURE 6.2.4 - 18

PENE. NOS. 71 PENETRATION PRESSURIZATION SYSTEM AIR SUPPLY



ENTIRE SYSTEM IS SEISMIC CLASS I

PENE. NOS. 67 CONTAINMENT CONTROLLED LEAK



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATION PRESSURIZATION SYSTEM AND CONTROLLED LEAK LINE	FIGURE 6.2.4 - 19

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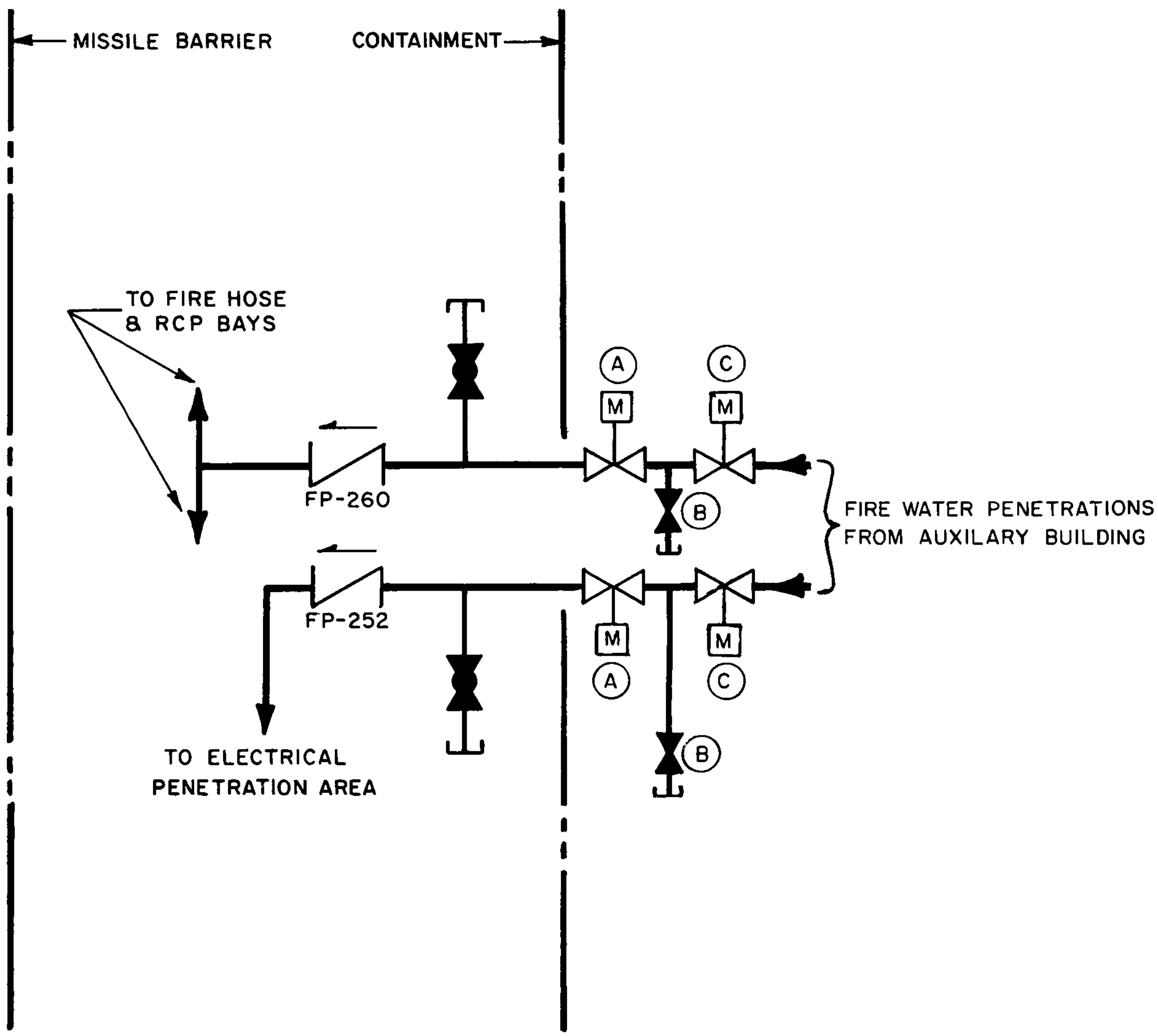
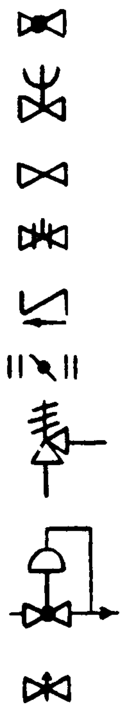


H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	FIRE WATER PENETRATIONS	FIGURE 6.2.4 - 20

LEGEND

VALVES

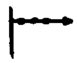


- GLOBE
- DIAPHRAGM
- GATE
- DOUBLE DISC GATE
- CHECK
- BUTTERFLY
- SAFETY OR RELIEF
- SELF CONTAINED PRESSURE REGULATOR
- NEEDLE

OPERATORS

- AIR DIAPHRAGM
- AIR CYLINDER
- MOTOR
- SOLENOID

DARKENED SYMBOL INDICATES NORMALLY CLOSED VALVE

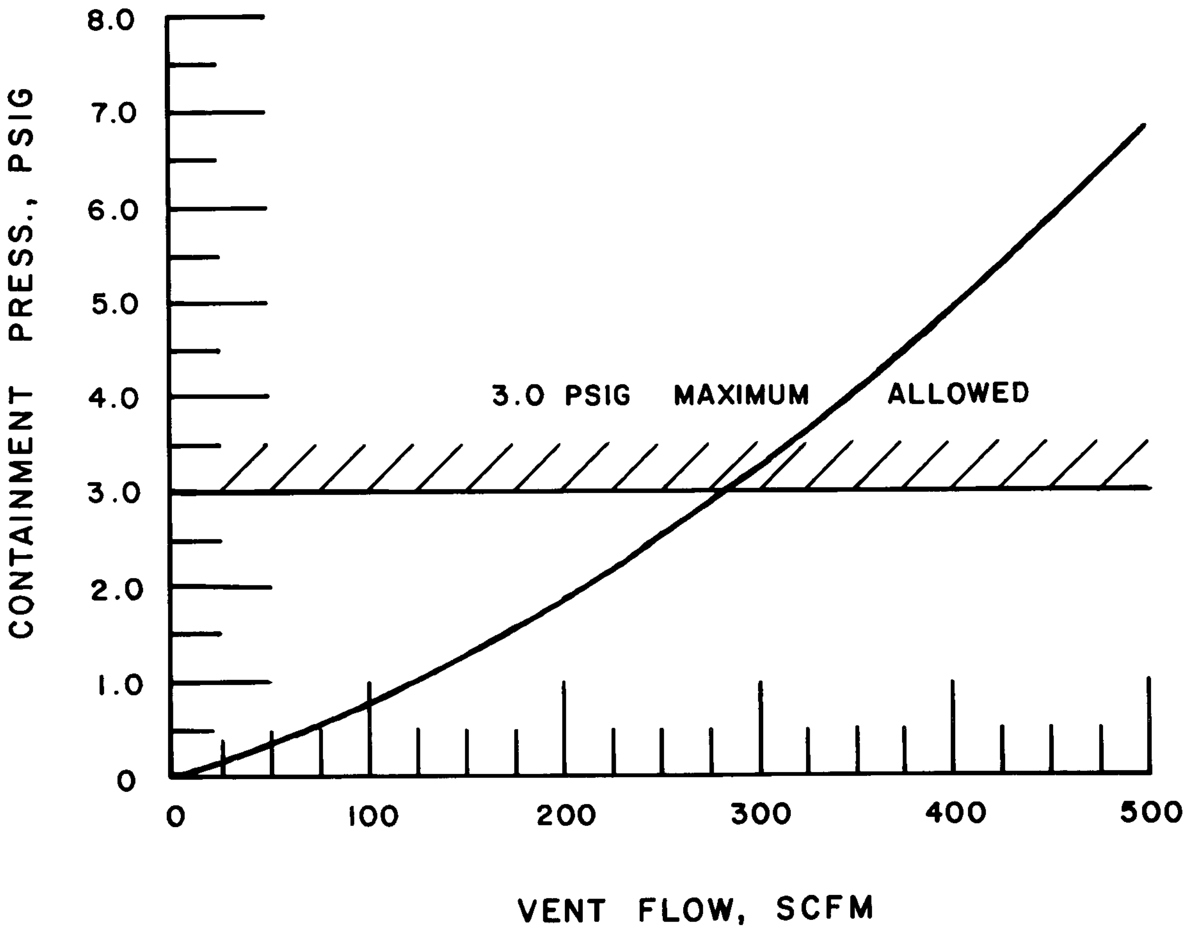


VALVE STEM LEAKOFF

NOTATION

T - TRIPPED BY CONTAINMENT ISOLATION SIGNAL, PHASE A
P - TRIPPED BY CONTAINMENT ISOLATION SIGNAL, PHASE B
IVSW (A) - ISOLATION VALVE SEAL WATER (AUTOMATIC)
IVSW (M) - ISOLATION VALVE SEAL WATER (MANUAL)
NO - NORMALLY OPEN
NC - NORMALLY CLOSED
FO - FAIL OPEN
FC - FAIL CLOSED
LO - LOCKED OPEN
LC - LOCKED CLOSED
PRT - PRESSURIZER RELIEF TANK
RWST - REFUELING WATER STORAGE TANK
DT - REACTOR COOLANT DRAIN TANK
S-I - SEISMIC CLASS I

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	LEGEND FOR SYMBOLS USED ON FIGURES 6.4.2 - 1 THRU - 20	FIGURE 6.2.4 - 21



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	POST-ACCIDENT VENTING SYSTEM SYSTEM RESISTANCE CURVE	FIGURE 6.2.5 - 2

When the break is large, RCS depressurization occurs due to the large rate of mass and energy loss through the break to the containment. The system is arranged so that the RHR pumps take suction from the sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is arranged to allow either of the RHR pumps to take over the recirculation function.

There are two sump return lines which lead from the containment to the RHR pumps. Each line is located inside of a larger diameter guard pipe. The lines are separated by approximately 18 ft. The lines are designed to allow for 2 in. differential movement between the containment and pump chamber and are designed as Seismic Class I equipment.

Filtration of the water entering the RHR pump suction piping during the recirculation mode is accomplished as follows: (Refer to Figure 6.3.2-3.)

a) Coarse filtration is accomplished by the screens in the lower portion of the shield wall. These screens have openings of approximately 1 in.

b) Floating and submerged debris is excluded from entering the pump suction by the baffles located in the sump area.

c) Any debris that penetrates the first two lines of defense (a. and b. above) is removed by the two screens (1/2 in. diameter and 7/32 in. diameter mesh) arranged in series at the pump suction line openings in the sump area.

Recirculation may start with a water depth of 1.5 ft on the containment floor. This is equivalent to the amount of water in the primary systems plus 60 percent of the RWST contents, or approximately 215,000 gal of water at 263°F. The maximum inlet velocity between the upper baffle and the container floor, which is the smallest flow area in this design, is approximately 1 ft/sec.

6.3.2.2.3 Net Positive Suction Head (NPSH) Requirements

The number of pumps operating, and the worst case flows for determining NPSH requirements are:

a) 3 high head pumps at 600 gpm each, or 1800 gpm total

b) 2 low head pumps at 3750 gpm each, or 7500 gpm total, and

c) 2 containment spray pumps at 1300 gpm each, or 2600 gpm total.

A quantitative analysis of the available and required NPSH for the SI, RHR and containment spray pumps for both the initial injection phase (with suction from the RWST) and the recirculation phase (suction from the containment sump) shows:

a) During the initial injection phase from RWST, (at initiation of this phase), the following applies:

Pump	NPSH, ft Required	Available
High head	25	51.1
Low head (RHR)	12	63.5
Containment spray	20	51.8

From this it can be seen that the high head pump is the controlling component for NPSH. The injection phase will be terminated just before the RWST level decreases to the point at which the available NPSH is reduced to the required NPSH of 25 ft at the runout flow of 600 gpm. Transition to recirculation from the containment sump will commence prior to this point.

b) During the recirculation phase (from containment sump) the following applies:

1) High head SI pumps - During recirculation via the high head pump, this pump and the RHR pump would be aligned in series, with the RHR pump (which has a head of 240 ft) boosting the suction of the high head pump. Thus, no NPSH problems would be experienced.

2) Containment spray pump - Same as high head SI pump.

3) RHR (low head) pump - During recirculation from the containment sump at 3750 gpm, the available NPSH with 2.5 ft of water on the containment floor is 20 ft. This takes credit for elevation head only. The required NPSH at 3750 gpm is 12 ft.

The high head recirculation flow path via the high head SI pumps is only required for the range of small break sizes for which the RCS pressure remains in excess of the shut-off head of the RHR pumps at the end of the injection phase.

Those portions of the SIS located outside of the containment which are designed to circulate, under post-accident conditions, radioactively contaminated water collected in the containment, meet the following requirements:

a) Shielding to maintain radiation levels within the guidelines set forth in 10CFR100

b) Collection of discharges from pressure relieving devices into closed systems, and

c) Means to limit radioactivity leakage to the environs, within guidelines set forth in 10CFR100.

Recirculation loop leakage is discussed in Section 6.3.2.5.5.

For the recirculation phase of the accident, the reactor coolant water which eventually is located on the containment floor is recirculated through the sump line from the containment to the suction of the RHR pump. Two independent and redundant recirculation lines are provided. Each line has two motor-operated valves. Both valves are located adjacent to the containment penetration in the RHR pit such that the line outside the containment can be isolated in the event of a passive failure. During recirculation, one recirculation train, which includes either of the two RHR pumps and either of the two residual heat exchangers, will be in service. The flow will go from the discharge of the RHR pump through the residual heat exchanger and then into the reactor via either the low head injection path or the high head injection path via the SI pumps. The high head injection path is provided in the event of a small break in which the pressure in the RCS is higher than the shut-off head of the RHR pumps.

In the event of a failure in the operating train during recirculation, the capability exists to switch to the other independent recirculation flow path; i.e., through the high head SI pumps to provide core cooling.

In the long term (post-accident) phase, injection through a separate header into the hot legs is possible by manual remote Control Room switch operation.

6.3.2.2.4 Cooling water

6.3.2.2.4.1 Component cooling system. During the recirculation mode, the Component Cooling System is used to cool the recirculation fluid as it passes through the residual heat exchanger. One of the three component cooling pumps and one of the two component cooling heat exchangers provide the cooling function during recirculation.

6.3.2.2.4.2 Service water system. The service water system is provided with redundant and independent loop headers and valves such that the two component cooling heat exchangers which are supplied with service water for cooling can have flow directed to them from the two independent headers. Two of the four service water pumps are required to operate during the recirculation phase.

6.3.2.2.5 Changeover from injection phase to recirculation phase. The sequence, from the time of the SI signal, for the changeover from the injection to the recirculation is as follows:

1. First, sufficient water is delivered into the containment during the injection phase to provide the required NPSH of the RHR pumps to allow the change to recirculation.

2. Second, the first low level alarm on the RWST sounds. At this point, the operator takes appropriate action to assure that sufficient NPSH exists for the operating pumps to run until the RWST is nearly empty. This alarm also serves to alert the operator to prepare for switchover to the recirculation mode.

3. Finally the second low level alarm on the RWST sounds. At this time, the operator performs the switchover operation.

The changeover from injection to recirculation is effected by the operator in the Control Room via a series of manual switching operations according to written procedures. Valves SI-856A and B are manually closed at the valves. Valve SI-870 A or B is manually closed if the motor operator is inoperable.

Remotely operated valves for the injection phase of the SIS (Figures 6.3.1-1 and 6.3.1-2) which are under manual control, (this is, valves which normally are in their ready position and do not receive a SI signal) have their positions indicated on a common portion of the control board. At any time during operation, when one of these valves is not in the ready position for injection, it is shown visually on the board. Table 6.3.2-1 is a listing of the instrumentation readouts on the control board which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

6.3.2.2.5.1 Location of the major components required for recirculation. The RHR pumps are located in the RHR pump pit (Elevation 203 ft 0 in.) which is below the basement floor of the Auxiliary Building (Elevation 226 ft 0 in). The RHR pump pit is located between the Containment Building and the Auxiliary Building. The residual heat exchangers are located on the first floor of the Auxiliary Building.

The high head SI pumps, component cooling pumps and component cooling heat exchangers are located in the Auxiliary Building (Elevation 226 ft 0 in).

The service water pumps are located in the intake structure, and the redundant piping to the component cooling heat exchangers is run underground.

6.3.2.2.6 Accumulators. The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation, each accumulator is isolated from the RCS by two check valves in series.

Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. Mechanical operation of the swing-disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The accumulators are passive engineered safety features (ESF) because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when the need arises. One accumulator is attached to each of the cold legs of the RCS.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop. The flow from the remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and one-half the core.

Valving is specified for exceptional tightness and, where possible, such as instrument valves, packless diaphragm valves are used. All valves except those which perform a control function are provided with backseats which are capable of limiting leakage to less than 1.0 cc/hr/in. of stem diameter, assuming no credit taken for valve packing. Those valves which are normally open are backseated. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves, 2 1/2 in. and above, which are exposed to recirculation flow, are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System or have had stuffing boxes live loaded and leakoff lines removed.

The check valves which isolate the SIS from the RCS are installed immediately adjacent to the reactor coolant piping to reduce the probability of an injection line rupture causing a LOCA.

Two relief valves are associated with the post loss-of-coolant recirculation. One is located outside the containment at the BIT discharge to prevent overpressure in the header and in the BIT. The high head SI piping leading to the hot legs is protected by a relief valve inside the containment in the test line.

The relieving capacity of these valves is based on a flow several times greater than the expected leakage rate through the check and isolation valves. They will also prevent overpressurization due to thermal expansion.

The RHR loop is protected by a relief valve in the common header leading to the accumulator pipes. The valve is located inside the containment and is relieved to the pressurizer relief tank. Apart from relieving possible leakage from the RCS, the valve is sized to relieve flow from one charging pump.

The gas relief valves on the accumulator protect them from pressures in excess of the design value.

6.3.2.2.12 Motor-operated valves. The pressure-containing parts (body, bonnet, and discs) of the valves employed in the SIS are designed per criteria established by the USAS B16.5 or MSS SP66 specifications. The materials of construction for these parts are procured to applicable ASME or ASTM specifications for austenitic stainless steel materials. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion resisting material. The pressure-containing cast components were radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body, bonnet, and discs were liquid penetrant inspected in accordance with the ASME Code, Section VIII, Appendix VIII. The liquid penetrant acceptable standard was as outlined in USAS B31.1 Case N-10.

When a gasket is employed, the body-to-bonnet joint was designed per ASME B&PV Code Section VIII or USAS B16.5 with a fully trapped, controlled compression, spiral wound, asbestos or graphite-filled gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194, respectively.

The entire assembled unit was hydrotested as outlined in MSS SP-61, with the exception that the test was maintained for a minimum period of 30 minutes per inch of wall thickness. Any leakage was cause for rejection. The seating design is of the Darling parallel disc design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear. Nickel-chrome-boron may be used as an alternate hard-surfacing material.

The stem material is ASTM A276 Type 316 condition B, or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse Specifications. These materials were selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. With the exception of valves which have been live loaded and have had leakoff lines capped, the valve stuffing box is designed with a lantern ring leak-off connection with a minimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1 1/2 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power plants.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to impact the discs away from the fore or backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed.

The valve was assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding and testing were submitted to Westinghouse for approval.

For those valves which must function on the SI signal, 10 sec operators are provided. For all other valves in the system, the valve operator completes its cycle from one position to the other within 120 sec.

Valves which must function against system pressure were designed such that they function with a pressure drop equal to full system pressure across the valve disc.

6.3.2.2.13 Manual valves. The stainless steel manual globe, gate, and check valves were designed and built in accordance with the requirements outlined in the motor-operated valve description above.

The carbon steel valves were built to conform with USAS B16.5. The materials of construction of the body, bonnet, and disc conformed to the requirements of ASTM A105 Grade II, A181 Grade II, or A216 Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and were subjected to hydrostatic test as outlined in MSS SP-61, except that the test pressure was

TABLE 6.3.2-5

PUMP PARAMETERS

Safety Injection Pump Design Parameters

Number	3
Design pressure, discharge, psig	1,750
Design temperature, °F	300
Design flow rate, gpm	375
Max. flow rate, gpm	550
Design head, ft	2,500
Shutoff head, ft	3,500
Material	11 - 13 Chrome
Motor H.P.	350
Type	Horizontal centrifugal

Residual Heat Removal Pump Design Parameters

Number of pumps	2
Type	Inline centrifugal
Design pressure, discharge, psig	600
Design temperature, °F	400
Design flow, gpm	3,750
Design head, ft	225
Material	Austenitic stainless steel
Motor H.P	300

TABLE 6.3.2-8 (Continued)

COMPONENT	MALFUNCTION	COMMENTS
3. High head cold leg isolation injection header valve (located at BIT inlet)	Fails to open	Two parallel valves. One required to open
4. Residual heat removal pump isolation valve at injection line	Fails to open	Two parallel valves. One required to open
D. Valves operated from Control Room for recirculation: (recirculation phase)		
1. Containment sump recirculation isolation	Fails to open	Two lines in parallel with two valves in series in each line, one pair of valves in either line is required to open
2. Safety injection pump suction valve at residual heat exchanger discharge	Fails to open	Two parallel valves. One required to open
3. Isolation valve on the test line returning to the refueling water storage tank	Fails to close	Two valve in series. One required to close. Both valves manually closed during switchover from injection to recirculation mode.
4. Isolation valve at suction header from refueling water storage tank	Fails to close	Two valves in series. One required to close

6.4.4 DESIGN EVALUATIONS

6.4.4.1 Radiological Protection

The evaluation of radiological exposures to plant operators from the DBA is presented in Chapter 15.

6.4.4.2 Toxic Gas Protection

The buildup of toxic chemical concentrations at the Control Room air intake and within the Control Room volume was evaluated to determine the effect on Control Room habitability from postulated toxic chemical releases.

Tables 6.4.4-1 and 6.4.4-2 summarize the general input data used in the analysis. Table 6.4.4-3 presents the identified offsite chemicals which were the subject of this analysis. No onsite toxic chemicals have been identified, excluding laboratory quantities of a few hazardous chemicals.

Concentrations of liquefied compressed gases at the Control Room air intake were analyzed using the procedures in Appendix B of Regulatory Guide 1.78. The quantity of the puff release (flash fraction) is evaluated assuming an isenthalpic expansion. Based on this analysis, chemicals with concentrations at the Control Room air intake less than the toxic limit were eliminated from further study.

Chemicals for which the calculated concentrations at the Control Room air intake exceed the toxic limit were analyzed further to determine the buildup of chemical concentration in the Control Room, using conservation of mass equations for the Control Room HVAC system operation. A Gaussian dispersion model was used to calculate the concentration dilution as the vapors drift from the spill site to the air intake. For purposes of this analysis, the normal mode HBR 2 plant ventilation system was used.

Table 6.4.4-2 summarizes the numerical results of this HBR 2 plant toxic chemical habitability analysis and shows compliance with the appropriate limits. The Regulatory Guide 1.78 screening procedure eliminated four chemicals stored in the vicinity as possible threats to Control Room habitability: propane, chlorine, nitrogen, and trichloroethylene. Of the remaining chemicals, only postulated releases of formaldehyde and ammonia result in calculated concentrations at the Control Room air intake that are greater than the toxic limit. The buildup in the Control Room was evaluated for these two chemicals, and the results are plotted in Figures 6.4.4-1 and 6.4.4-2. While the peak toxic chemical concentration exceeds the recommended limit for habitability, both ammonia and formaldehyde are readily detectable by odor and eye irritation. The odor thresholds are shown in Figures 6.4.4-1 and 6.4.4-2. The elapsed time between reaching the odor threshold and the toxic limit concentrations in the Control Room for both chemicals is seen in these figures to be greater than the 2 min time allowed in Regulatory Guide 1.78 for operators to locate and put on self-contained breathing apparatus. Two self-contained breathing apparatuses are available in the Control Room for use in emergencies.

TABLE 6.4.4-3

RESULTS OF TOXIC CHEMICAL ANALYSIS

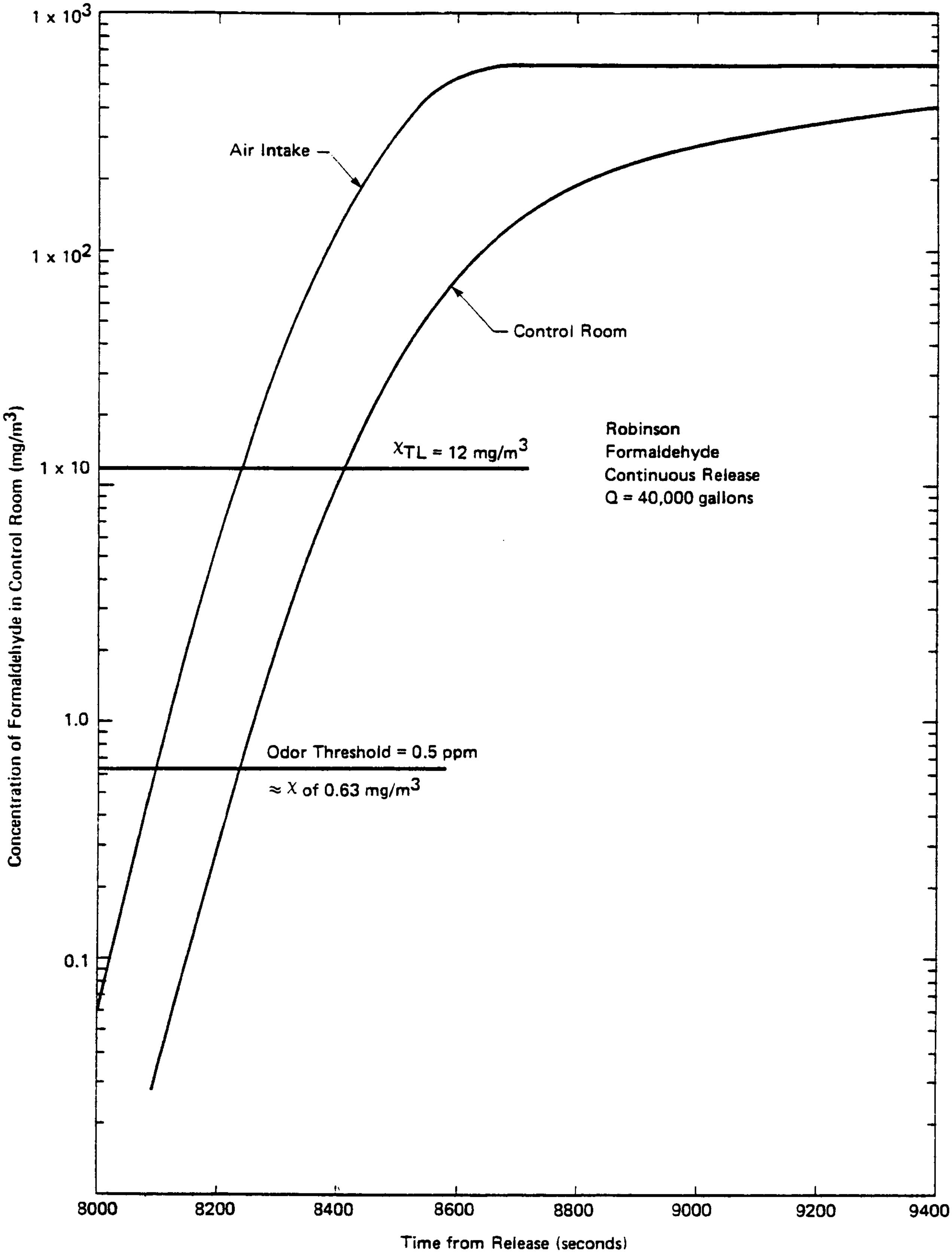
LOCATION	DISTANCE (mi)	CHEMICAL	STORAGE CONDITION	QUANTITY STORED (1000 gal)	QUANTITY STORED (lb)	QUANTITY ALLOWED PER REGULATORY GUIDE 1.78 (lb)	TOXIC LIMIT (mg/m^3)	CONCENTRATION AT INTAKE (mg/m^3)	CONCENTRATION IN CONTROL ROOM AT 2 MIN AFTER HUMAN DETECTION (mg/m^3)
Sonoco	5.1	Propane	Liquefied gas under pressure, 2 tanks	10	48.8K	235K	1800	N/C[a]	N/C
Sonoco	5.3	Sulfuric acid	Ambient temperature and pressure	25	382.5K	262	2	0.08	N/C
Sonoco	5.4	Phenol	Ambient temperature and pressure	10	89.4K	2487	19	3.7	N/C
Sonoco	5.4	Acetic acid	Ambient temperature and pressure	10	87.5K	3270	25	18	N/C
Sonoco	5.4	Formaldehyde (aqueous)	Ambient temperature	40	334K (123.5K)[b]	1570	12	996	<12[c]
Sonoco	5.3	Chlorine	Liquefied gas under pressure	<1.0	12.3K (2840)[d]	5890	45	N/C	N/C
IMC	5.0	Ammonia (anhydrous)	Liquefied gas under pressure	--	160K (36.5K)[d]	9160	70	3120	<70[e]
IMC	5.0	Nitrogen	---	--	184K	7.85M	Asphyxiant, 84%	N/C	N/C
IMC	4.8	Sulfuric acid	Ambient temperature and pressure	--	200K	262	2	0.077	N/C

TABLE 6.4.4-3 (Cont'd)

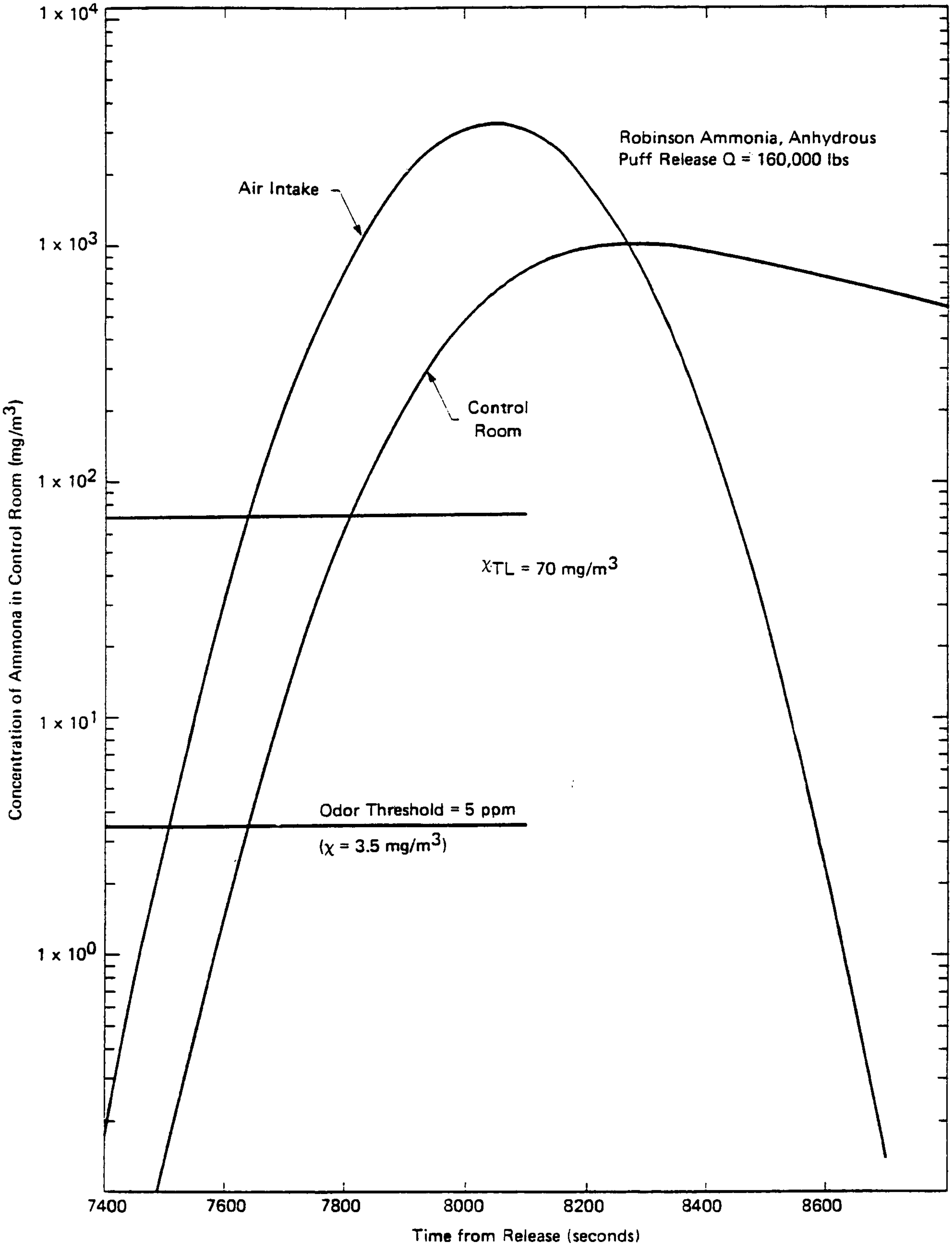
RESULTS OF TOXIC CHEMICAL ANALYSIS

LOCATION	DISTANCE (mi)	CHEMICAL	STORAGE CONDITION	QUANTITY STORED (1000 gal)	QUANTITY STORED (lb)	QUANTITY ALLOWED PER REGULATORY GUIDE 1.78 (lb)	TOXIC LIMIT (mg/m^3)	CONCENTRATION AT INTAKE (mg/m^3)	CONCENTRATION IN CONTROL ROOM AT 2 MIN AFTER HUMAN DETECTION (mg/m^3)
Hartsville Mill	5.3	Trichloro-ethylene	55 gal drum, ambient temperature and pressure	--	668	1570	535	N/C	N/C
Darlington County Water Well Heads	Various, all more than 100 meters from Control Room	Chlorine	Liquefied gas under pressure	--	150	(f)	45	N/C	N/C

[a]N/C = Not required to be calculated.
[b]Amount of formaldehyde gas in standard industrial formaldehyde solution.
[c]See Figure 6-2 for a plot of Control Room formaldehyde concentration as a function of time.
[d]Isenthalpic flash fraction.
[e]See Figure 6-3 for a plot of Control Room ammonia concentration as a function of time.
[f]According to Regulatory Guide 1.95, all chlorine must be at least 100 meters from the Control Room. The four Hartsville Public Works Wells in question are scattered over Darlington County at various distances (miles) from the plant, much more than 100 meters away.



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTROL ROOM CONCENTRATION AFTER POSTULATED RELEASE OF FORMALDAHYDE	FIGURE 6.4.4 - 1



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTROL ROOM CONCENTRATION AFTER POSTULATED RELEASE OF AMMONIA	FIGURE 6.4.4 - 2

6.8.2 System Design

6.8.2.1 System Description. The IVSW system flow diagram is shown in Figure 6.8.2-1.

System operation is initiated either manually or by any automatic safety injection (SI) signal. When actuated, the IVSW System interposes water inside the penetrating line between two isolation points located outside the containment. The resulting water seal blocks leakage of the containment through valve seats and stem packing. The water is introduced at a pressure slightly higher (approximately 47 psig) than the containment design pressure of 42 psig. The possibility of leakage from the containment or the RCS past the first isolation point is thus prevented by assuring that if leakage does exist, it will be from the seal water system into the containment. Service is discontinued after the manual reset buttons for PCV-1922 A and B are reset after a containment isolation Phase A reset.

The system includes one seal water tank capable of supplying the total requirements of the system. The tank is pressurized with a nitrogen blanket supplied from two independent sources. Primary supply is from the plant nitrogen supply header through a pressure regulating control valve. Automatic backup supply is provided from two high pressure nitrogen bottles through separate high and low pressure regulating valves. Design pressure of the tank and piping is 150 psig. The injection piping runs and the piping from the nitrogen supply bottles are fabricated using 3/8 in. OD stainless steel tubing, which is capable of 2500 psig service. Relief valves are provided to prevent over-pressurization of the system if a pressure control valve fails, or if a seal water injection line communicates with a high pressure line due to a check valve failure in the seal water line. The seal water tank requires no external power source to maintain the required driving pressure.

Local instrumentation is also provided, as shown in Figure 6.8.2-1. The primary source of N_2 from the plant N_2 supply header is backed up by two, independent, high pressure N_2 bottles. If there should be a break or failure of the N_2 header, the N_2 blanket pressure is maintained by the tanks and blowdown through the N_2 header is prevented by check valves.

The tank supplies pressurized water to four distribution headers. Header "A" is the manual header, meaning an isolation valve on this header must be pressurized by opening a manual valve supplying the individual isolation valve. Headers "B", "C", and "D" are automatic headers that are pressurized through one or both of two redundant, fail-open, air-operated valves in parallel. These valves open on receipt of an SI signal. A loss of power will cause the automatic valves to open, since automatic initiation is a de-energized signal to vent air from the valve operators. System operation is initiated by a Phase A containment isolation signal which accompanies any SI signal. System operation is discontinued after the manual reset buttons for valves PCV-1922 A and B are reset after the Phase A reset.

Liquid carrying piping two inches and larger with design pressure or temperature exceeding 200 psig or 200°F is isolated by one manual or remote-operated, double disc gate valve. A drawing of this valve is presented in Figure 6.8.2-2. Redundant isolation barriers are provided when the valve is closed. The upstream and downstream discs are forced against their respective seats by the closing action of the valve. Seal water is injected through thevalve bonnet or body and pressurizes the space between the two valve discs.

The seal water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is provided by two globe valves in series with the seal water injected into the pipe between the valves. The valves are oriented such that the seal water wets the stem packing. When the valves are closed for containment isolation, the first isolation point is the valve plug in the valve closest to containment, and the water seal is applied between the valve plug and stem packing. In a number of the smaller lines, isolation is provided by two diaphragm (Saunders Patent) valves in series, with the seal water injected into the pipe between the valves.

The maximum acceptable leakage across both the seat and stem packing of any gate or globe valve is 10 cc/hr/in. of nominal pipe diameter. Tests on these valves have indicated that much lower leakage rates can be expected. However, the design of the IVSW System is based on the conservative assumption that all isolation valves are leaking at five times the acceptable value, or 50 cc/hr/in. of nominal pipe diameter.

In addition, should one of the isolation valves fail to close, flow through the failed valve will be limited by a restricting orifice to a maximum leakage value of 63,200 cc/hr. A water seal at the failed valve is assured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

The seal water tank is sized to provide at least a 24 hr supply of seal water under the following adverse circumstances: isolation valves leaking at the design rate of 50 cc/hr/in. plus the failure of the largest containment isolation valve to seat, resulting in leakage at the maximum rate of 1000 cc/hr/in. The seal water tank is sized to satisfy these conditions. Two separate, independent, seismically qualified sources of makeup water (primary water and service water) are provided to ensure that an adequate supply of seal water is available for long-term operation. Service water makeup is from two sources - the service water header, and from each of the service water booster pumps. This assures a redundant long-term supply of water from a source at greater than the 1.1 times the design pressure (approximately 46 psig). Based on maximum leakage and flows into the tank from makeup sources, use of the makeup source would be required for only minimal amounts of time each day at very low flows which will not affect other functions of the makeup system.

<u>6.8.2.2 Isolation Valve Seal Water Actuation Criteria</u>. Containment isolation and seal water injection are accomplished automatically for certain penetrating lines requiring early isolation, and manually for others, depending on the status of the system being isolated and the potential for leakage in each case.

The automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic SI actuation, and trips the majority of the automatic isolation valves. These valves are in the so-called "non-essential" process lines penetrating the containment. This is defined as "Phase A" isolation, and the trip valves are designated by the letter "T".

This signal also initiates automatic seal water injection. The second, or "Phase B" containment isolation signal, is derived upon actuation of the Containment Spray System, and trips the automatic isolation valves in the so-called "essential" process lines penetrating the containment. These trip valves are designated by the letter "P".

A manual containment isolation signal or SI signal can be generated from the Control Room. This signal performs the same functions as the automatically derived "T" signal, i.e., "Phase A" isolation and automatic seal water injection.

Generally, the following criterion determines whether the isolation and seal water injection is automatic or manual. Automatic containment isolation and automatic seal water injection are required for lines that could communicate with the containment atmosphere and be void of water following a LOCA.

These lines include: 6

a) Reactor coolant pump seal water return line (Phase B isolation)

b) Letdown line

c) RCS sample lines

d) Reactor coolant vent line

e) Containment air sample inlet and outlet lines (air pressurization), and

f) Reactor coolant drain tank gas analyzer line.

Automatic containment isolation and automatic seal water injection are also provided for the following lines, which are not connected directly to the RCS, but terminate inside the containment at certain components. These components can be exposed to the reactor coolant or to the containment atmosphere as the result of leakage or failure of a related line or component. The isolation lines are not required for post-accident service.

These lines include:

a) Pressurizer relief tank gas analyzer line

b) Pressurizer relief tank makeup line

c) SI System test line

d) Reactor coolant drain tank pump discharge line

e) Steam generator blowdown lines

f) Steam generator blowdown sample lines

g) Accumulator sample line, and

h) Containment sump pump discharge.

Manual containment isolation and manual seal water injection are provided for lines that are normally filled with water and will remain filled following the LOCA, and for lines that must remain in service for a time following the accident. The manual seal water injection assures a long-term seal. These lines include:

a) Reactor coolant pump seal water supply lines

b) Charging line

c) SI headers

d) Boron injection lines, and

e) Containment spray headers.

Seal water injection is not necessary to ensure the integrity of isolated lines in the following categories:

a) Lines that are connected to non-radioactive systems outside the containment, and in which a pressure gradient exists that opposes leakage from the containment. These include nitrogen supply lines to the pressurizer relief tank, accumulators, the reactor coolant drain tank, the instrument air header, the pressurizer deadweight tester line, and the plant air header.

b) Lines that do not communicate with the containment atmosphere or RCS and are missile-protected throughout their length inside containment. These lines are not postulated to be severed or otherwise opened to the containment atmosphere as a result of a LOCA. These include the steam and feedwater headers, the containment ventilation system cooling water supply and return lines, and the excess letdown heat exchanger cooling water supply and return lines.

c) Lines that are designed for long-term, post-accident service as part of the engineered safety features. The only lines in this category are the containment sump recirculation lines. These lines are connected to a closed system outside containment.

d) Special lines such as the fuel transfer tube, containment purge ducts, and the containment pressure and vacuum relief lines. The zone between the two gaskets sealing the blind flange to the inner end of the fuel transfer tube is pressurized by the Penetration Pressurization System (PPS) to prevent leakage from the containment in the event of an accident. The zone between the two butterfly valves in each containment purge duct is pressurized above incident pressure by the PPS while the valves are closed during power operation, as are the spaces between the two butterfly valves in the containment pressure and vacuum relief lines.

7.0 INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

The HBR 2 instrumentation and controls, including both safety-related and nonsafety-related systems, are described in this chapter. Systems described are the Reactor Protection System (RPS), systems which initiate engineered safeguards, instrumentation and controls required for safe shutdown, display instrumentation, and control systems not required for safety.

The RPS monitors all parameters related to safe operation of the reactor. The system is designed to trip the reactor so as to protect the core against fuel rod cladding damage caused by departure from nucleate boiling (DNB), and to protect the Reactor Coolant System (RCS) against damage caused by over-pressure. The Engineered Safety Features (ESF) Instrumentation System monitors parameters to detect failure of the RCS, and initiates containment isolation and ESF operation.

The RPS automatically trips the reactor to protect the reactor core when the following conditions exist:

a) The reactor power, as measured by neutron flux, reaches a preset limit

b) The temperature rise across the core as determined from loop ΔT reaches a limit, either from a fixed ΔT setpoint (function of T_{avg} and neutron flux distribution) or a variable ΔT setpoint (function of T_{avg}, pressurizer pressure, and neutron flux distribution)

c) The pressurizer pressure reaches a minimum limit, or

d) There is a loss of reactor coolant flow as sensed by low flow, loss of pump power or pump circuit breakers opening.

The RPS automatically trips the reactor to protect the RCS when the pressurizer pressure or level reaches a maximum limit. There are also trips on safety injection, turbine-generator trip, steam/feedwater flow mismatch, low-low steam generator level, source range neutron flux, and intermediate range neutron flux.

Interlocking functions of the RPS prevent control rod withdrawal when a specified parameter reaches a value less than the value at which reactor trip is initiated.

For anticipated abnormal conditions, protective systems in conjunction with inherent plant characteristics and ESF are designed to assure that limits for energy release to the containment and for radiation exposure are not exceeded.

The ESF actuation system automatically performs the following vital functions:

a) Starts operation of the Safety Injection System (SIS) upon low pressurizer pressure, or high containment pressure, or high differential pressure between any steam line and the steam line header, or on high steam flow in any two steam lines, coincident with low steam pressure or low reactor coolant average temperature

b) Operates the Phase A containment isolation valves upon automatic actuation of the SIS

c) Starts the Containment Spray System and operates the remaining containment isolation valves upon detection of a high-high containment pressure signal

d) Starts operation of the Containment Air Recirculation System if not operating after operation of the required SIS is initiated

e) Closes all steam line isolation valves on occurrence of any of the following:

 1) High steam flow in any two steam lines coincident with low steam pressure or low reactor coolant average temperature

5 2) High-High containment pressure signals

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 instrumentation and control systems were designed in accordance with guidance available at the time. This included proposed General Design Criteria (discussed in Section 3.1) and IEEE-279 (proposed, 1968, discussed in Section 7.2).

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.

BF_3 gas filled 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. 4

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
4 detector for each source range channel is a BF_3 gas filled 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.
4 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.

TABLE 7.2.1-2

PERMISSIVE CIRCUITS

NUMBER	FUNCTION	REQUIRED INPUT
1	Prevent rod withdrawal on overpower	1/4 high neutron flux (power range) or 1/2 high neutron flux (intermediate range) or 2/3 over-temperature ΔT or 2/3 overpower ΔT
2	Auto-rod withdrawal stop at low powers	Low Mwe (15% power) load signal (turbine pressure)
3	Auto-rod withdrawal stop on rod drop	1/4 rapid decrease of neutron flux (power range) or 1/1 rod bottom indication
4*		
5	Steam dump interlocks	Rapid decrease of Mwe load signal (turbine pressure)
6	Manual block of source range trip	1/2 high intermediate range flux allows manual block, 2/2 low intermediate range defeats block
7	Permissive power (block various trips). Required only at power	3/4 low-low neutron flux (power range) and 1/2 low Mwe load signal (turbine pressure)
8	Block single primary loop loss of flow trip	3/4 low neutron flux (power range)
9*		
10	Manual block of low power range trip (power range) intermediate range trip	2/4 high neutron flux allows manual block, 3/4 low neutron flux (power range) defeats manual block

*Not applicable to this plant.

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. 8

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, with the exception of the opening of both feedwater pump circuit breakers and AMSAC. 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

6) AMSAC trip (two of three SG below low-low setpoint at $\geq$ 40% power)

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 2/2 logic to indicate a loss of voltage on any one bus.

3) Manually

4) AMSAC trip (two of three SG below low-low setpoint at $\geq$ 40% power)

In the Loss of Normal Feedwater analysis, in Section 15.2.7, 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.

To prevent the start of the auxiliary feedwater pumps under shutdown conditions, key switches have been installed as discussed in Section 7.3.2.2.2.

6. 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. SI-856A and B fail to the preferred position during normal operation but fail to the unpreferred position with a manual handwheel as backup during the recirculation mode of operation. 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.

7. 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.

8. 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.

9. 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

10. 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.

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

12. RHR Pit Level Indication and Alarms - Level indication has been provided on the RTGB to monitor water level in the RHR pits. This indication in conjunction with corresponding HI and HI-HI RHR Pit A&B Level Annunciation on APP-001 will provide operators with sufficient time to isolate any water sources to the particular RHR pump pit to preclude a common mode RHR pump failure.

7.3.1.1.2.4 Interlocks to prevent diesel generator overload during safety injection.

1. 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.

2. Pressurizer Heaters - The 150 kW group of pressurizer heaters used to assure natural circulation at hot standby conditions are fed from redundant diesel generator buses during loss of off-site power. However, upon initiation of a safety injection signal, the pressurizer heater load will be shed to prevent overloading its DG.

3. 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 approximately 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 7.2.1.3.

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.

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 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

7.4.1 DESCRIPTION

The Control Room Building, its equipment, and furnishings have been designed so that the likelihood of fire or other conditions which could render the Control Room inaccessible even for a short time is extremely small.

As a further measure to assure safety, provisions have been made so that plant operators can shut down and maintain the plant in a safe condition by means of controls located outside the Control Room. During such a period of Control Room inaccessibility, 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, via the refueling water storage supply. The capability to achieve and maintain cold shutdown conditions from outside the Control Room, in the event of a fire, is also provided.

Local controls are located so that the stations to be manned and the times when attention is needed are within the capability of the plant operating crew. The plant intercom system provides communication among the personnel so that the operation can be coordinated.

For a description of the systems required for safe shutdown in the event of a fire, refer to Appendix 9.5.1C.

The functions for which local control provisions have been made are listed below along with the type of control and its location in the plant. Transfer to these local controls is annunciated in the Control Room.

7.4.1.1 Equipment Control Outside Control Room

7.4.1.1.1 Reactor Shutdown

If the Control Room should be evacuated suddenly without any action by the operators, the reactor can be tripped by either of the following:

a) Open rod control breakers at the reactor trip switchgear

b) Actuate the manual turbine trip

When the reactor is 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 1 percent reactivity shutdown margin provided by the full length control rods had been cancelled. This delay would provide time for useful emergency measures.

7.4.1.1.2 Residual Heat Removal

Failure to maintain water supply to the steam generators following a normal plant shutdown would result in steam generator dry out after some 400 sec and loss of the secondary system for decay heat removal. 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. Numerous calculations, verified by startup tests on the Connecticut-Yankee and San Onofre Power Plants have shown that with the steam generator safety valves operating alone, the RCS maintains itself close to the nominal no-load condition. The steam relief facility is adequately protected by redundancy and local protection.

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 motor driven auxiliary feed pumps or by the steam-driven auxiliary feed pump. In addition to the normal feed circuit the plant may fall back on:

a) The condensate storage tanks

b) Service water

c) Onsite deep well water

7.4.1.1.3 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 (CVCS). This requirement implies operation of a charging pump. In the event that all charging pumps are inoperable (as the result of a fire that causes damage to cables serving all three pumps), RCS inventory can be maintained by depressurizing the RCS and utilizing the safety injection pumps.

7.4.1.1.4 Indication and Controls Provided Outside the Control Room

The specific indication and controls provided outside the Control Room for the above capability are summarized as follows:

a) Indication

1) Level Indication for the Individual Steam Generators

(a) Motor Driven Auxiliary feed pumps room
(b) Main feed bypass control valves
(c) Turbine Building mezzanine dedicated shutdown (DS) panel
(d) Charging pump room DS panel

2) Pressure Indication For the Individual Steam Generators

(a) Turbine Building mezzanine DS panel

7.4.2 ANALYSIS

7.4.2.1 Compliance With Applicable Codes and Standards

The engineered safety feature systems were designed in accordance with the applicable General Design Criteria (GDC) effective in 1968. The reactor protection system was also designed in accordance with applicable GDC and IEEE 279, "Proposed Criteria for Nuclear Power Plant Protection Systems," August, 1968. No regulatory guides were available for incorporation into the original design criteria for the engineered safety features.

The alternative/dedicated shutdown system modifications do not impact the physical integrity of the auxiliary shutdown system components. The only penetrations into the existing system pressure boundary were for the installation of new impulse lines for new DS system instrumentation. The modification did not impact the system process and the auxiliary shutdown system will continue to meet all of the original mechanical and operational design criteria.

The post-fire safe-shutdown modifications provide for:

a) Separation of Redundant Circuits

Where safety-related circuits have been modified, new wiring and components have been installed so that, as a minimum, the separation requirements of Regulatory Guide 1.75 are met. The basis for the modifications (fire hazards analysis) dictated that power and control wiring for selected components (e.g., one charging pump, one service water pump) be rerouted so that cables serving redundant pumps would not pass through common fire areas.

b) Fault Isolation for Safety-Related Circuits and Power Supplies

Electrical isolation, in accordance with Regulatory Guide 1.75, is provided to ensure that external faults (fire-induced) will not degrade existing or new safety-related electrical systems.

c) Separation of Safety and Non-Safety Related Circuits

Isolation devices and/or physical separation are provided to ensure that failures in non-safety related circuits will not jeopardize adjacent safety-related circuits.

d) Annunciation in Main Control Room or Bypass or Assumption of Local Control

For those components provided with a "control transfer" features, auxiliary contacts on each control transfer switch are used to provide annunciation (in the Control Room) when the component is switched out of its "remote control" mode.

This annunciation feature has been implemented for all auxiliary shutdown components having remote/local control capabilities.

e) Interlocks and Administrative Controls to Limit the Consequence of Faulted Conditions

Features such as key interlocks or racking out of selected circuit breakers prevent the inadvertent cross-connection or simultaneous faulting of redundant power supplies.

f) Seismic Installation in Safety-Related Areas or Safety-Related Cabinets

Interfaces with existing safety-related cabinets and new safety-related cabinets (e.g., charging pump room panel, transfer switch panels) and their included components have been designed to remain functional through a safe shutdown earthquake (SSE).

g) Single-Failure Criterion

All new safety-related components and safety-related interfaces are designed so that a single failure cannot cause the loss of redundant safety systems. The modifications generally affect only one of redundant equipment trains. The failure of one of these equipment trains will not initiate the failure of the redundant train; electrical and physical separation of the redundant trains have not been degraded as a result of the modification.

7.4.2.2 Control and Power Circuit Separation

In conducting the fire hazard analysis, it was determined that several plant fire areas were critical in that cables for redundant shutdown-related components were routed through these areas. As a result, a severe fire in one of these areas could incapacitate redundant equipment trains by destroying power and control cables, or by destroying power supplies.

In order to mitigate the consequences of a fire in any one of the plant fire areas, power and control cables for selected shutdown-related components were rerouted to avoid these areas. The alternate power sources and local control panels and the device itself are independent of the areas of concern or a means of manual or remote operation has been provided.

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7.4.2.3 Operating Requirements

7.4.2.3.1 Operators

Sufficient qualified operators will be available to conduct shutdown activities, whether from the control room or following a control room evacuation. Operator staffing is adequate for implementation of post-fire shutdown procedures in the event of a fire in any plant fire area.

7.4.2.3.2 Equipment Adequacy

The functions required for post-fire shutdown are equivalent to those functions that must be maintained, as a minimum, in a safe-shutdown scenario that does not involve accident mitigation functions. The specific equipment credited to perform these functions, and the adequacy of the equipment to fulfill the necessary operational objectives are discussed in Appendix 9.5.1C.

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REFERENCES: SECTION 7.4

7.4.1-1 Safe Shutdown Component/Cable Separation Analysis [10CFR50, Appendix R, Section III.G] for H. B. Robinson Unit 2.