WM. H. ZIMMER POWER STATION

INSTRUCTIONS FOR UPDATING YOUR FSAR

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TABLE 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

(Parameters are Related to Rated Power Output
for a Single Plant unless Otherwise Noted)

THERMAL AND HYDRAULIC DESIGN	ZPS-1	HATCH 1	BAILLY 1*	LA SALLE 1 & 2
Rated power, MWt	2436	2436	1931	3293
Design power, MWt (ECCS** design basis)	2550	2550	2028	3489
Steam flow rate, lb/hr	10.477×10^6	10.03×10^6	7.925×10^6	14.166×10^6
Core coolant flow rate, lb/hr	78.5×10^6	78.5×10^6	61.5×10^6	106.5×10^6
Feedwater flow rate, lb/hr	10.477×10^6	10.445×10^6	7.925×10^6	14.127×10^6
System pressure, nominal in steam dome, psia	1020	1020	1020	1020
Average power density, kW/liter	50.50	51.21	51	49.15
Maximum thermal output, kW/ft	13.4	18.5	18.5	13.4
Average thermal output, kW/ft	5.45	7.10	7.113	5.34
Maximum heat flux, Btu/hr-ft^2	354,000	428,300	428,240	361,600
Average heat flux, Btu/hr-ft^2	144,100	164,400	164,707	144,100

* The Bailly application has been withdrawn. However, the original Bailly system parameters have been retained in the Section 1.3 tables for historical purposes.

** Emergency core cooling system.

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TABLE 1.3-1 (Cont'd)

THERMAL AND HYDRAULIC DESIGN (Cont'd)	ZPS-1	HATCH 1	BAILLY 1	LA SALLE 1 & 2	
Maximum UO_2 temperature, °F	3336	4493	4380	3435	68
Average volumetric fuel temperature, °F	2130	2781	1100	2184	82
Average fuel rod surface temperature, °F	566	565	558	567	
Minimum critical heat flux ratio (MCHFR)	1.21*	≥1.9	≥1.9	1.28	
Coolant enthalpy at core inlet, Btu/lb	527.4	526.2	527.1	527.1	38
Core maximum exit voids within assemblies	75	79	78	76	
Core average exit quality, % steam	13.6	12.9	13.6	13.2	
Feedwater temperature, °F	420	387.4	420	420	
Design Power Peaking Factor					
Maximum relative assembly power	1.4	1.4	1.4	1.4	
Local peaking factor	1.24	1.24	1.24	1.15	68
Axial peaking factor	1.4	1.5	1.5	1.4	38
Total peaking factor	2.43	2.6	2.6	2.25	68

* Minimum critical power ratio (MCPR) value; GETAB used in ZPS-1 FSAR 82

TABLE 1.3-1 (Cont'd)

CORE MECHANICAL DESIGN	ZPS-1	HATCH 1	BAILLY 1	LA SALLE 1 & 2
Fuel Assembly				
Number of fuel assemblies	560	560	444	764
Fuel rod array	8x8	7x7	8x8	8x8
Overall dimensions, in.	176.16	176.16	175.83	176.16
Weight of UO_2 per assembly, lb	465.15	466.9	Undished - 490.35 Dished - 483.42	458.0*
Weight of fuel assembly, lb	697.72	674.7	Undished - 681.48 Dished - 674.55	600.0
Fuel Rods				
Number per fuel assembly	63	49	63	62
Outside diameter, in.	0.493	0.563	0.563	0.483
Cladding thickness, in.	0.034	0.037	0.032	0.032
Gap, pellet to cladding, in.	0.009	0.012	0.006	0.009
Length of gas plenum, in.	14	16	16	9
Cladding material	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2
Cladding process	Free-standing loaded tubes	Free-standing loaded tubes	Free-standing loaded tubes	Free-standing loaded tubes

*Pellets (flat ends and chamfered)

38 82 38 82 68 1 38 82 38

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TABLE 1.3-1 (Cont'd)

CORE MECHANICAL DESIGN (Cont'd)	ZPS-1	HATCH 1	BAILLY 1	LA SALLE 1 & 2	
Fuel Pellets					
Material	UO_2	UO_2	UO_2	UO_2	
Diameter, in.	0.416	0.477	0.487	0.410	38 | 82
Length, in.	0.420	0.5	0.5	0.410	
Fuel Channel					
Overall dimension, length, in.	166.906	166.875	166.906	166.906	
Thickness, in.	0.100	0.080	0.080	0.100	38 ZPS-1
Cross section dimensions, in.	5.478x5.478	5.438x5.438	5.438x5.438	5.478 x 5.478	
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	
Core Assembly					
Fuel weight as UO_2, lb	260,484	261,452	226,350	349,912	38 | 82
Core diameter (equivalent), in.	160.2	160.2	142.1	187.1	
Core height (active fuel), in.	146	144	144	150	38

TABLE 2.5-28 (Cont'd)

STRUCTURE	OCTOBER 1981 ELEVATION (ft MSL)	OCTOBER 1981 SETTLEMENT (ft) (in.)	JANUARY 1982 ELEVATION (ft MSL)	JANUARY 1982 SETTLEMENT (ft) (in.)
Reactor Building				
	520.939	0.109 1.31	520.936	0.112 1.34
	506.227	0.035 0.42	506.224	0.038 0.46
Auxiliary Building				
	506.514	0.087 1.04	506.515	0.086 1.03
Turbine Building				
	520.941	0.069 0.83	520.934	0.076 0.91
	520.950	0.060 0.72	520.942	0.068 0.82
	506.046	0.062 0.74	506.057	0.051 0.61
Heater Bay				
	520.360	0.078 0.94	520.961	0.077 0.92
Circulatory Water Pump Structure				
	520.487	0.070 0.84	[illegible]	0.065 0.78
	--	-- --	--	-- --
	520.077	0.199 2.39	[illegible]	0.189 2.27
	520.278	0.200 2.40	[illegible]	0.185 2.22
Diesel Generator				
	520.937	0.073 0.88	520.935	0.075 0.90
Radwaste Building				
	521.498	0.108 1.30	521.499	0.107 1.28
	521.296	0.108 1.29	521.295	0.109 1.31
Machine and Service Building				
	519.480	0.095 1.14	519.481	0.094 1.13
Storage Building				
	--	-- --	--	-- --
	--	-- --	--	-- --
	520.4[illegible]	0.063 0.76	520.455	0.068 0.82
Service Building Addition				
	520.392	0.009 0.11	520.388	0.013 0.16

fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

3.1.2.4.12.1 Evaluation Against Criterion 41 | 75

As described in other sections of this FSAR (Section 9.4), ventilation and refrigeration systems are provided in the drywell and reactor building areas to maintain suitable temperature conditions and to provide thorough mixing of the atmospheres during normal operation.

The standby gas treatment system (SGTS) is utilized during abnormal conditions to maintain the reactor building at a negative pressure and to filter the exhaust air of potential fission products. Recirculation fans and duct works are provided to mix the reactor building air to provide a dilute sample to the SGTS equipment trains for filtering and release to the atmosphere. | 82

The drywell and suppression chamber purge system is provided to clean up the drywell and suppression chamber atmospheres prior to entry of personnel during normal operation. | 79

A combustible gas control system is provided to control the concentration of combustible gas in the primary containment. The system has redundant active components and power supplies that conform to the requirements of Regulatory Guide 1.7 (draft Regulatory Guide dated September 1976). The containment atmosphere is continuously monitored for combustible gas concentration, and the control system can be manually operated as required. | 78

The above described systems meet the requirements of Criterion 41. For further discussion, see the following chapters, sections, or subsections:

a. General Plant Description, Section 1.2;

b. Containment Functional Design, in Subsection 6.2.1;

c. Containment Air Cleanup, in Subsection 6.2.3;

d. Instrumentation and Controls, Chapter 7.0;

c. Design Limits

1. Design Local Strain - The permanent strain in metallic ductile material is limited to 50% of the minimum actual uniform elongation based on restraint material tests.

2. Design Load - Due to dynamic loading and strain hardening effects, the static yield strength of the material is increased by 15%. Tensile yielding members are designed to yield at this higher value.

3.6.3.2.5.1.2 Compressive Deforming Members

a. Materials

1. standard hexagonal core aluminum honeycomb manufactured according to standard specification MIL-C-7438F (February 1977), or | 82

2. facing materials at each end of the honeycomb assembly are hot-rolled carbon steel, ASTM A36-70a, or approved equal. Attachment of the facing material to the honeycomb is by a bonding adhesive or brazing, either of which is capable of withstanding the temperature and loading conditions.

b. Performance Requirements

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1. the material has a usable (minimum) axial strain of 50%,

2. the crushable energy-absorbing material will not need to be replaced when subjected to an overall accident temperature of 340° F,

3. the material remains functional under an operating gamma radiation dosage of 1.8×10^7 rads and neutron fluence of 1.8×10^{14} neutrons/cm^2, and an accident gamma radiation dosage of 2.6×10^7 rads, and

4. the honeycomb is precrushed to ensure uniform resistence with displacement.

c. Design Limits

1. The function of the crushable energy-absorbing material is to absorb the pipe whip energy at a constant load during a pipe-break accident.

2. The axial strain in the honeycomb is limited to 50%.

3.6.3.2.5.2 Type II

a. Materials - Plates, bars, and shapes conformed to the applicable requirements of one of the following ASTM Standards,

3.6.5.2.2 Jet Impingement Shields

Jet impingement shields such as the sacrificial shield doors are provided to limit the consequences of rupture of the piping and are designed to withstand the resultant jet forces using the codes specified in Section 3.8. 82

3.6.5.3 Separation

Maintaining the independence of redundant safety systems and components is achieved in most cases by separating the redundant components so that no single postulated event can prevent the safety-related function from occurring. This is achieved by the following:

a. physical separation of source and target;

b. routing of cables so that different penetrations and paths are utilized to ensure that one event will not preclude both the primary and backup components from fulfilling their design function;

c. deflection utilized to redirect a jet spray from an essential component;

d. utilization of intermediate components and structure to intercept and defray forces; and

e. location of duplicate instrument lines to ensure that one cause will not preclude each of the redundant systems from fulfilling their design function.

3.6.5.4 Typical Pipe Whip Restraints

The typical pipe whip restraint used on the reactor recirculation piping system is shown in Figure 3.6-26. In this type of restraint, steel bars are used for energy absorbers in tension only.

There are four types of pipe whip restraints used on the remaining high energy systems inside containment which are shown in Figure 3.6-27. The specific type used at a specific location depends on the surrounding geometry of the structure and the most probable direction of loading.

The first type of pipe whip restraint (Figure 3.6-27(a)) is the basic three bar restraint. The two outer bars are comprised of necked-down steel plates designed to yield and carry load only in tension. The middle bar is referred to as the compression post.

When the most probable direction of loading is to load the compression post, the second type of pipe whip restraint is utilized (Figure 3.6-27(b)). This restraint is like the first type except that there is crushable material on the end of the compression post used to absorb energy.

Where space does not permit using type 1 or 2, type 3 (Figure 3.6-27(c)) is utilized. In this type of restraint the two outer bars of the type 1

systems and components which are sensitive to dynamic seismic events.

3.7.2.1.2.7.1 Dynamic Analysis of Piping Systems

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as changes in stiffness due to curved members. Next the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the piping system is being anchored and supported at points with different excitations, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple support excitation analysis methods may be used when acceleration time-histories or response spectra appropriate to each are applied to each of the individual piping system attachment points in the structural model. The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor point displacement are used for a static analysis to determine the additional stresses due to relative anchor point displacements. 82 73

The preliminary dynamic model of the recirculation piping and the steamline piping systems are given in Figures 3.7-9 and 3.7-10. The final model will be supplied after completion of the stress report.

3.7.2.1.2.7.2 Dynamic Analysis of Equipment

Equipment is idealized as a mathematical model consisting of lumped masses connected by elastic members or springs. Results for some selected large Seismic Category I equipment are given in Table 3.7-3.

Seismic loadings due to two orthogonal horizontal directions and the vertical are combined as detailed in Subsection 3.7.2.1.2.4.

When the equipment is supported at more than two points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple support excitation analysis methods may be used when acceleration time-histories or response spectra appropriate to each are applied to each of the individual equipment attachment points.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support point displacements are used for a static analysis to determine the additional stresses due to support displacements. Further details are given in Subsection 3.7.2.1.2.11.

3.7.2.1.2.8 Seismic Qualification by Testing

For certain Seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results will reflect the following:

a. performance data of equipment, which under the specified conditions has been subjected to dynamic loads equal to or greater than those to be experienced under the specified seismic conditions,

The postulated pipe rupture at the weld between recirculation or feedwater piping and the reactor nozzle safe-end leads to a high flow rate of water and steam mixture into the annulus between the RPV and the shield wall. Calculation of the mass/energy release is performed using the generic method for short term mass releases. The GE annulus pressurization mass energy release rate calculation was compared to Relap 4, MOD 5, which is the NRC criteria, and found acceptable with the GE criteria more conservative than the NRC criteria.

3.9.1.1.3.1 Acoustic Loads

Because the Boiling Water Reactor (BWR) is a two-phase system that operates at or close to saturation pressure (1000 psi), the differential pressure across the reactor shroud is of short duration and the BWR system is not subjected to a significant shock type load with respect to structural supports. This short duration acoustic load is confined to a bending moment and shear force on the reactor pressure vessel and reactor shroud support.

3.9.1.1.3.2 Pressure Loads

The pressure responses of the RPV-shield wall annulus for a recirculation suction line postulated rupture and a feedwater line postulated rupture were investigated using the RELAP 4 computer code. An asymmetric model, using several nodes and flow paths, was developed for the analysis of the recirculation and feedwater line ruptures. Further description of these analytical models and detailed discussion of the analysis may be found in FSAR Section 6.2. The methods used to calculate the pressure distribution within the annulus were described in detail in Subsection 5.3.1.4 of the ZPS Mark II Design Assessment Report (DAR), Amendment 13, October 1980 and FSAR Subsection 3.9.1.1.3. Sensitivity studies were performed which confirmed that the methods used were conservative. 49 82

3.9.1.1.3.3 Jet Loads

Structural loads on the vessel and internals, jet thrust, jet impingement and pipewhip restraint loads were considered in conjunction with the above mentioned pressure loads. Jet thrust refers to the vessel reaction force which results as the jet stream of liquid is released from the rupture. Jet impingement refers to the jet stream force which leaves the broken pipe and impacts the vessel. The pipe-whip restraint load is the force which results when the energy absorbing pipe-whip restraint restricts the pipe separation to less than one full pipe diameter. These jet loads are calculated as described in Reference 2.

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific components. "The GE computer programs are maintained either by General Electric or by outside computer program developers. In either case, the quality of the programs and the computed results are controlled. For each program, one or more individuals are assigned. Their duties are

a. to keep abreast of the capability, the software contents, and the theory of the program,

wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer holddown brackets attached to the reactor vessel top head. 82

3.9.5.1.10 Feedwater Spargers

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is welded at each feedwater thermal sleeve which is welded to the vessel nozzle and is shaped to conform to the curve of the vessel wall.

Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.11 Core Spray Lines

The core spray lines are the means for directing flow to the core spray nozzles inside the shroud which distribute coolant so that peak fuel cladding temperatures of 2200° F are not exceeded during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles. The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger, which is routed halfway around the inside of the upper shroud. The ends of the two spargers are supported by brackets designed to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and vessel. The other core spray line is identical except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the spargers. 49

3.9.5.1.12 Vessel Head Cooling Spray Nozzle

When reactor coolant is returned to the reactor vessel, part of the flow can be diverted to a spray nozzle in the reactor head. This spray maintains saturated conditions in a reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The spray also decreases thermal stratification in the reactor vessel coolant. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and therefore limits thermal stress in the vessel during cooldown.

The vessel head cooling spray nozzle is flange mounted to a mating flange on the reactor vessel head nozzle.

TABLE 3.9-21

REACTOR REFUELING & SERVICING EQUIPMENT

NEW FUEL STORAGE RACK

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
A. Aluminum Casting ASTM B26 SG70A-T51 (1972)				
Fu = 23,000 psi Fy = 16,000 psi				
Normal Conditions S_{limit} = 0.66Fy	Normal Operating Loads	Axial Load Plus Bending	10,560	3,421
Emergency Condition S_{limit} = 0.88 Fy (Upset limit)	Normal Operating Loads, Operating Basis Earthquake, Safety Relief Valve and LOCA	Axial Load Plus Bending	14,080 (Upset operation limit)	9,912
Faulted Condition S_{limit} = the lesser of 1.2 Fy and 0.7 Fu	Normal Operating loads, Safe Shutdown Earthquake, Safety Relief Valve and LOCA	Axial Load Plus Bending	16,100	15,536
B. Aluminum Casting ASTM B108 SG70A-T6 (1972)				
Fu = 33,000 psi Fy = 22,000 psi				

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TABLE 3.9-24 (Cont'd)

Notes:

1. Operability demonstrated by analysis.
2. The stresses calculated for the faulted operating condition loads are less than the allowable stress limits for the upset condition.
3. This component has been evaluated using dynamic loads unique to ZPS-1. The governing loading combinations listed in Table 3.9-2 have been applied appropriately. The stresses calculated in this evaluation are less than the stresses calculated from the original design-basis loads.

a. required to detect a steamline accident condition;

b. required to perform a steamline isolation function; or

c. required to perform a water line isolation function and could be subjected to the steam environment, such as electrical cable or valve operator.

Rotating machinery (pumps, motors, operators) in a safety system with a leak is designed to function in the local environment caused by the leak, if the equipment is specified in a. through c. preceding.

All balance-of-plant engineered safety features equipment is designed to operate in the accident environment applicable to the location of the equipment as designated in Tables 3.11-2 and 3.11-3. Class 1E equipment is qualified by testing or by analysis in accordance with procedure set forth in IEEE 323-1971. The procurement specification for the Class 1E equipment stipulates IEEE 323-1971 standard. The minimum environmental design requirements for the Class 1E equipment are specified in the procurement specification. Components sensitive to extreme environmental conditions are located in areas served by safety related cooling systems, such as the control room, the electrical equipment room, and the switchgear rooms. No separate environmental enclosures are needed to house balance-of-plant engineered safety features equipment. 12

3.11.3 Qualification Test Results

Table 3.10-1 summarizes the normal and accident qualification environments to which Class 1E equipment was exposed during qualification testing. Detailed test results for Class 1E equipment by General Electric are on file at GE - San Jose. 12

Seismic test results for Class 1E equipment not supplied by General Electric are on file at Sargent & Lundy and at the ZPS-1 site. 14

3.11.4 Loss of Ventilation

3.11.4.1 Control Room Air-Conditioning and Ventilation System

Controls and electrical equipment necessary for safe plant shutdown are located in the control room. The control room is air-conditioned and shielded against radiation to allow the operators safe and continued occupancy under optimum environmental conditions. Air-conditioning equipment and associated components and the refrigeration equipment condensers are designed to Seismic Category I requirements. 13 82

The above non-destructive test methods are in compliance with the following ASTM Specifications:

a. E94 (1968) Radiographic Testing;

b. E142 (1968) controlling Quality of Radiographic Testing;

c. E114 (1971) Ultrasonic Testing by the Reflection Method Using Pulsed Longitudinal Waves Induced by Direct Inspection;

d. E164 (1968) Ultrasonic Contact Inspection of Weldments; 77 78

e. E109 (1971) Magnetic Particle Inspection; Dry Powder:

f. E138 (1971) Magnetic Particle Inspection; Wet; and

g. E165 (1971) Liquid Penetrant Inspection.

3.13.4 Containment Liner and Other Steel Elements Serving Pressure Vessel Functions

3.13.4.1 Materials

The materials essential to containment integrity comply with the requirements of the 1971 ASME Boiler and Pressure Vessel Code, Winter '71 addenda, Section III, Paragraph NE-2300, and comply to the following specifications:

APPLICATION	SPECIFICATION
Equipment Hatch and Personnel Lock	SA516-1970a, Grade 70
Containment Liner	SA516-1970a, Grade 60
Containment Liner Anchors	A36-1970a
Pipe Penetrations	Pipe A333-1970a, Grade I Rolled Plate, SA5.6-1970a, Grade 60, Butt-welded 78
Drywell Head	SA516-1970a, Grade 70
Immersion Area of Suppression Chamber	SA358, Class I, Grade 304-1966 or SA240, type 304-1970 82

The liner plate welds are examined as described in these paragraphs to meet the requirements of NRC Regulatory Guide 1.19 (Revision 1). 75 77

Charpy V-notch tests are done at a temperature of at least 30° F below the lowest service metal temperature. In addition, the drywell floor embedment ring (Figure 3.8-4) is tested for laminations in accordance with the applicable requirements of ASTM A435 (1967). 77

TABLE 3.13-1 (Cont'd)

ASTM DESIGNATION	TITLE	
E329-1970	Recommended Practice for Inspection and Testing Agencies for Concrete and Steel as Used in Construction	81
D75-1968	Sampling Aggregates	81

AASHO DESIGNATION		
T26-1972	Standard Method of Test for Quality of Water to be Used in Concrete.	82

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82

required. There are also other tests, such as the cold hydro tests and the hot functional tests to be performed. Pre-operational tests on each system will be performed on site.

The valves are also seismically qualified using, as minimum, the peak acceleration evaluated at the valve center of gravity as input. The seismic qualification procedure for valves is described in Subsections 3.7.3.16, 3.9.2.2, and 3.9.3.3. | 77

With the loads known from above, the structural analyses were performed with other conservative loads to meet the stress criteria. This will ensure that the critical parts of the concerned component will not be damaged during and after the faulted condition. | 77

Finally, active valves are also required to be operated periodically as specified in the Technical Specification. This repeated operability requirement throughout the life of the specified valve further provides a complete operability assurance program.

The list of RCPB Class 1 active valves is given in Table 5.2-5. The list of ASME III (1971 through Winter 1972 Addenda), Classes 2 and 3 pumps and valves is given in Table 3.9-25. | 2 | 77 | 82

The representative combination of loads and analysis to ensure operability is summarized in Table 5.2-3B. | 2

5.2.1.8 Inadvertent Operation of Valves

A discussion of the design-basis events and their appropriate limits for this plant is given in Chapter 15.0. The events in Chapter 15.0 have been selected to envelope the most severe change in critical parameters from events which have been postulated to occur during planned operation.

5.2.1.9 Stress and Pressure Limits

Paragraphs NB-3655 and NB-3656 (Piping Sections) of ASME Section III (1971 through Winter 1972 Addenda) are not directly applicable to pumps and valves. On the basis of the utilized method of establishing system design pressures, however, it can be stated that the permissible pressure requirements of NB-3655.1 and NB-3556.1 are met. | 77

The allowable stress limits and design loads based on applicable codes for RCPB components are summarized in Table 5.2-3B. Active or inactive components of the RCPB are delineated in Table 5.2-5. | 2

5.2.1.10 Stress Analysis for Structural Adequacy

Stress analysis was used to determine structural adequacy of pressure components of the reactor coolant pressure boundary under various operating conditions and earthquakes.

2. RHR shutdown cooling isolation valves,

3. RHR discharge to radwaste,

4. RHR head spray,

5. drywell equipment and floor drain sump isolation valves, 68

6. containment monitoring sample lines, and

7. drywell and suppression pool purge;

h. isolate RR flow control valve hydraulic lines;

i. isolate trip the drywell chilled water;

j. initiate the suppression pool spray timer; 68

k. initiate the wetwell spray timers; and 43 82

l. initiate post-LOCA containment monitoring system.

The drywell pressure will continue to increase at a rate dependent upon the size of the assumed steam leak. The largest instrument line is only 1-inch diameter (neglecting the restriction orifice) and has an internal area of 0.005 ft^2. The pressure increase in the drywell will depress the water level in the suppression pool vents until the level reaches the bottom of the vents. At this time, air and steam will start to enter the suppression pool. The steam will be condensed and the air will pass to the suppression chamber free air space. The latter will result in a gradual pressurization of the containment dependent upon the air carryover rate. Assuming no bypass leakage, the peak suppression pool pressure will not exceed 35 psig. 27

The significantly reduced rate of steam generation following a reactor scram causes the turbine control valves to close in an attempt to maintain reactor pressure. The generator antimotoring protection or closure of the turbine control valves trips the main turbine and generator. Reactor vessel pressure continues to decrease through the steam leak and steam demand from station auxiliaries. When reactor pressure reaches 840 psig, the main steam isolation valves are automatically closed.

At this point, decay heat will cause pressure in the vessel to increase and start opening the relief valves at approximately 1070 psig using the pneumatic operator function of the relief valve. Reactor pressure remains relatively constant at this valve for the first 10 minutes by automatic actuation of the relief valves. Steam from the vessel is dumped to the suppression pool through the relief valves, causing reactor water level to decrease.

The high-pressure core spray (HPCS) system, initiated when drywell pressure reached +2 psig, starts spraying into the vessel approximately 30 seconds after the high drywell pressure signal is initiated and continues until the vessel level reaches +58 inches. At this level the HPCS injection valve closes automatically; however, the HPCS pump continues to operate through the minimum flow bypass line ready to spray again should level decrease to -38 inches.

6.2.5.4 Testing and Inspections

Each active component of the combustible gas control system is testable during normal reactor power operation.

Preoperational tests of the combustible gas control system are conducted during the final stages of plant construction prior to initial startup (see Chapter 14.0). These tests assure correct functioning of all controls, instrumentation, recombiners, piping,and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the preoperational tests and are used as base points for measurements in subsequent operational tests. 82

6.2.5.5 Instrumentation Requirements

See Section 7.3.

6.2.5.6 Materials

Generally, the same materials are used in the combustible gas control system as in the other containment systems (see Subsection 6.2.2.6 and 6.2.4.5 for example). At present, it appears that the only material unique to the combustible gas control system will be an Inconel 800 tube in the recombiner system. This tube has been analyzed, as has all other material in the combustible gas control system, to assure that it will not radiolytically or pyrolytically decompose in the atmosphere to which it might be exposed. Therefore, no decomposition products will be formed which might interfere with the operation of this or any other engineered safeguard system.

6.2.5.7 Containment Inerting

Section 50.44 of 10 CFR 50 requires containment inerting of Mark II containments. The equipment and piping to inert the ZPS-1 containment will be provided before plant startup. 82

Figures 6.2-64 and 9.4-7 show the inerting system piping and equipment. An outdoor station that includes the liquid nitrogen storage tanks, vaporizers, and control stat-ion will supply nitrogen gas to the containment. The high containment purge flow via the 3-inch line to the suppression pool and the 4-inch line to the drywell will reduce the containment oxygen content to 4% or less within 8 hours during the plant startup operation. The makeup flow via the 1-½-inch lines to the suppression pool and drywell will maintain the containment during all reactor operating periods to less than 4% oxygen.

TABLE 6.

ISOLATION VALVE SUMMARY FOR LINE

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION				VA
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN	
M.S. Drain	1B21F016	M-7	I.C.	55	Gate MO	Auto	RM	Open	Open	
M.S. Drain	1B21F019	M-7	O.C.	55	Gate MO	Auto	RM	Open	Open	
Spare	--	M-8	--	--	--	--	--	--	--	
RBCCW	1WR074	M-9	I.C.	57	Gate AO	RM	--	Open	Open	Ope
RBCCW	1WR055	M-9	O.C.	57	Gate MO	RM	Man.	Open	Open	Ope
RHR Head Spray	1E51F066	M-10	I.C.	55	Check	Rev. Flow	--	Shut	Shut	
RHR Head Spray	1E51F317	M-10	I.C.	55	Globe MO	RM	Man.	Shut	Shut	
RHR Head Spray	1E12F023	M-10	O.C.	55	Globe MO	Auto	RM	Shut	Open	
RHR LPCI	1E12F042B	M-11	O.C.	55	Gate MO	RM	Man.	Shut	Shut	
RHR LPCI	1E12F042C	M-12	O.C.	55	Gate MO	RM	Man.	Shut	Shut	
RHR LPCI	1E12F042A	M-13	O.C.	55	Gate MO	RM	Man.	Shut	Shut	
CRD Return	1C11F383	M-14	I.C.	55	Check	Rev. Flow	--	Shut	Shut	
CRD Return	1C11F382	M-14	O.C.	55	Check	Rev. Flow	--	Shut	Shut	
CRD Return	1C11F082	M-14	O.C.	55	Gate MO	RM	Man.	Shut	Shut	
LPCS to RPV	1E21F005	M-15	O.C.	55	Gate MO	RM	Man.	Shut	Shut	
HPCS to RPV	1E22F004	M-16	O.C.	55	Gate MO	RM	Man.	Shut	Shut	
RHR Drywell Spray	1E12F017B	M-17	O.C.	56	Gate MO	RM	Man.	Shut	Shut	
RHR Drywell Spray	1E12F016B	M-17	O.C.	56	Gate MO	RM	Man.	Shut	Shut	
RHR Steam	1E51F063	M-18	I.C.	55	Gate MO	Auto	RM	Open	Open	
RHR Steam	1E51F076	M-18	I.C.	55	Globe MO	RM	Man.	Shut	Shut	
RHR Steam	1E51F064	M-18	O.C.	55	Gate MO	Auto	RM	Shut	Shut	

E POSITION		CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
OST LOCA	POWER FAILURE										
	As is	B,C,D,E,F,J	3	STD.	Yes	A	1	Y	No	See Notes 11, 18	
	As is	B,C,D,E,F,J	3	STD.	Yes	A	1	Y	No	See Note 11	76
	--	--	--	--	--	--	--	--	--	Type "A" Test	
Shut	Shut	X	6	STD.	No	C	1	Y	Yes		
Shut	As is		6	STD.	No	B	1	Y	Yes		76
	--	--	4	--	No	A	1	Y	No		
	As is	X --	3/4	STD.	No	B	1	Y	No		77
											79 73
	As is	A,G,S,T	6	STD.	Yes	A	1	Y	No		76 82
	As is	--	10	STD.	Yes	A	1	Y	Yes	See Note 18	
	As is	--	10	STD.	Yes	A	1	Y	Yes	See Note 18	
	As is	--	10	STD.	Yes	A	1	Y	Yes	See Note 18	
	--	--	3	--	No	A	1	Y	No		
	--	--	3	--	No	A	1	Y	No		
	As is	--	3	STD.	No	A	1	Y	No		
	As is	--	10	STD.	Yes	A	1	Y	Yes	See Note 18	
	As is	--	10	STD.	Yes	A	1	Y	Yes	See Note 18	
	As is	A,L	14	70	Yes	B	1	Y	Yes	See Notes 8a, 11, 18	
	As is	A,L	14	70	Yes	B	1	Y	Yes	See Notes 8a, 11	
	As is	R,S,V_1,V_2,V_3,V_4,V_5	8	18	Yes	A	1	Y	No	See Notes 12, 18	76
	As is	--	1	STD.	No	A	1	Y	No	See Notes 12, 18	
	As is	R,S,V_3,V_4	8	18	Yes	A	1	Y	No	See Note 12	76 82

TABLE 6.2

ISOLATION VALVE SUMMARY FOR LINES

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION				VA
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN	
Spare		M-19	--	--	--	--	--	--	--	
RHR Shutdown Suct.	1E12F009	M-20	I.C.	55	Gate MO	Auto	RM	Shut	Open	S
RHR Shutdown Suct.	1E12F008	M-20	O.C.	55	Gate MO	Auto	RM	Shut	Open	S
RHR Drywell Spray	1E12F017A	M-21	O.C.	56	Gate MO	RM	Man.	Shut	Shut	S
RHR Drywell Spray	1E12F016A	M-21	O.C.	56	Gate MO	RM	Man.	Shut	Shut	S
Spare	--	M-22	--	--	--	--	--	--	--	
RBCCW	1WR054	M-23	O.C.	57	Gate MO	RM	Man.	Open	Open	Ope
RBCCW	1WR073	M-23	I.C.	57	Gate AO	RM	Man.	Open	Open	Ope
RCIC Steam	1E51F063	M-24	I.C.	55	Gate MO	Auto	RM	Open	Open	S
RCIC Steam	1E51F008	M-24	O.C.	55	Gate MO	Auto	RM	Open	Open	S
RHR Shutdown Retrn	1E12F053B	M-25	O.C.	55	Globe MO	Auto	RM	Shut	Open	S
RHR Shutdown Retrn	1E12F053A	M-26	O.C.	55	Globe MO	Auto	RM	Shut	Open	S
RT	1G33F001	M-27	I.C.	55	Gate MO	Auto	RM	Open	Open	S
RT	1G33F004	M-27	O.C.	55	Gate MO	Auto	RM	Open	Open	S
TIP System	1C51J004	M-28	O.C.	54	Ball SO	Auto	Man.	Shut	Shut	S
(A,B,C,D)	1C51J004	M-28	O.C.	54	Shear SO	RM	--	Shut	Shut	S
TIP Purge	1IN161	M-28	O.C.	54	Check	Rev. Flow	--	Open	Open	S
TIP Purge	1C51J009	M-28	O.C.	54	Gate SO	Auto	Man.	Open	Open	S
CRD Hydr. Insert.		M-29	O.C.	55	Gate	Man.	NA	Open	Open	Op
CRD Hydr. Withdr.		M-29	O.C.	55	Gate	Man.	NA	Open	Open	Op
Equipment Hatch	--	M-30	--	--	--	--	--	--	--	
Personnel Hatch	--	M-31	--	--	--	--	--	--	--	

VE POSITION: POST LOCA	VE POSITION: POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
	--	--	--	--	--	--	--	--	--	Type "A" Test	
t	As is	A,G,S,T	18	36	Yes	A	1	Y	Yes		
t	As is	A,G,S,T	18	36	Yes	A	1	Y	Yes		73
t	As is	A,L	14	70	Yes	B	1	Y	Yes	See Notes 8a, 11, 18	
t	As is	A,L	14	70	Yes	B	1	Y	Yes	See Notes 8a, 11	
	--	--	--	--	--	--	--	--	--	Type "A" Test	
Shut	As is	--	6	STD.	No	B	1	Y	Yes		76
Shut	Shut	X	6	STD.	No	C	1	Y	Yes		
											82
t	As is	R,S,V_1,V_2,V_3,V_4,V_5	8	30	Yes	A	1	Y	No	See Note 12	76
t	As is	V_1,V_2,V_5	4	20	Yes	A	1	Y	No	See Note 12	
t	As is	A,G,S,T	10	36	Yes	A	1	Y	Yes	See Note 18	
t	As is	A,G,S,T	10	36	Yes	A	1	Y	Yes	See Note 18	
t	As is	B,N,K	6	30	Yes	A	1	Y	No	See Note 18	82
t	As is	B,N,K,W	6	30	Yes	A	1	Y	No		76
t	As is	X	3/8	STD.	No	NA	1	Y	No	TYP for A,B,C,&D	
t	Shut	X	3/8	STD.	No	NA	1	N	No	See Notes 19, 16 TYP for A,B,C,&D	
t	--	--	1/2	--	No	NA	1	Y	No		
t	Shut	A,L	1/2	STD.	No	NA	1	Y	No		
en	NA	None	1	NA	No	B	1	N	Yes	See Note 9	76
en	NA	None	3/4	NA	No	B	1	N	Yes	See Note 9	
-	--	--	--	--	--	--	--	--	--	Type "B" Test	
-	--	--	--	--	--	--	--	--	--	Type "B" Test	

TABLE

ISOLATION VALVE SUMMARY FOR LI

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION			
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDO
Access Hatch	--	M-32	--	--	--	--	--	--	--
Access Hatch	--	M-33	--	--	--	--	--	--	--
RHR "B" Suction	1E12F004B	M-34	O.C.	56	Gate MO	RM	Man.	Open	Shut
RHR "C" Suction	1E12F004C	M-35	O.C.	56	Gate MO	RM	Man.	Open	Open
RHR "A" Suction	1E12F004A	M-36	O.C.	56	Gate MO	RM	Man.	Open	Shut
LPCS Suction	1E21F001	M-37	O.C.	56	Gate MO	RM	Man.	Open	Open
HPCS Suction	1E22F015	M-38	O.C.	56	Gate MO	RM	Man.	Shut	Open
RCIC Turbine Exht.	1E51F068	M-39	O.C.	56	Gate MO	RM	Man.	Open	Open
RCIC Exhaust V.B.	1E51F080	M-39	O.C.	56	Globe MO	Auto	RM	Open	Open
RCIC Exhaust	1E51F344	M-39	O.C.	56	Check	Rev. Flow	--	Shut	Shut
RCIC Vac. Pmp. Disc	1E51F069	M-40	O.C.	56	Globe MO	RM	Man.	Open	Open
RCIC Vac. PMP. Disc	1E51F028	M-40	O.C.	56	Check	Rev. Flow	--	Shut	Shut
RCIC Pmp. Suct.	1E51F031	M-41	O.C.	56	Gate MO	RM	Man.	Shut	Shut
RCIC Min. Flow	1E51F019	M-42	O.C.	56	Globe MO	RM	Man.	Shut	Shut
RHR "A" Wet Well Spray	1E12F027A	M-43	O.C.	56	Gate MO	Auto	RM	Shut	Shut
LPCS Test	1E21F012	M-44	O.C.	56	Globe MO	Auto	RM	Shut	Shut
RHR "A" Hx. Cond.	1E12F011A	M-44	O.C.	56	Gate MO	Auto	RM	Shut	Shut
LPCS Min. Flow	1E21F011	M-44	O.C.	56	Gate MO	Auto	RM	Shut	Shut
RHR "A" Min. Flow	1E12F064A	M-44	O.C.	56	Gate MO	RM	Man.	Shut	Shut

VALVE POSITION POST LOCA	VALVE POSITION POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	RESULTS	
--	--	--	--	--	--	--	--	--	--	Type "B" Test	
--	--	--	--	--	--	--	--	--	--	Type "B" Test	
Open	As is	--	20	STD.	Yes	B	1	Y	Yes	See Note 18	
Open	As is	--	20	STD.	Yes	B	1	Y	Yes	See Note 18	
Open	As is	--	20	STD.	Yes	B	1	Y	Yes	See Note 18	
Open	As is	--	20	STD.	Yes	B	1	Y	Yes	See Note 18	
Open	As is	--	20	STD.	Yes	B	1	Y	Yes		73
pen/Shut	As is	--	8	STD.	No	B	1	Y	No	See Notes 26, 18	
pen/Shut	As is	L,V4,	2	STD.	Yes	B	1	Y	No	See Notes 26, 18	
pen/Shut	--	--	8	STD.	No	B	1	Y	No	See Note 26	79
pen/Shut	As is	--	2	STD.	No	B	1	Y	No	See Notes 26, 18	
pen/Shut	--	--	2	STD.	No	B	1	Y	No	See Note 26	
Open	As is	--	6	STD.	No	B	1	Y	No	See Note 18	
pen/Shut	As is	X	2	STD.	No	B	1	Y	No	See Notes 18, 26	82
Shut	As is	U,L	4	STD.	Yes	B	1	Y	Yes	See Notes 8B, 18	
Shut	As is	U,L	12	STD.	Yes	B	1	Y	Yes	See Note 18	
Shut	As is	A,L,X	4	STD.	Yes	B	1	Y	Yes	See Note 18	76
pen/Shut	As is	X	4	STD.	Yes	B	1	Y	Yes	See Note 18	
pen/Shut	As is	X	4	STD.	Yes	B	1	Y	Yes	See Note 18	

TABLE 6.

ISOLATION VALVE SUMMARY FOR LIN

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION		V	
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN
RR "B" Hyd. Contl.	1B33F356A	M-69	O.C.	57	Globe SO	Auto	RM	Open	Open
RR "B" Hyd. Contl.	1B33F356C	M-69	O.C.	57	Globe SO	Auto	RM	Open	Open
RR "B" Hyd. Contl.	1B33F356B	M-69	O.C.	57	Globe SO	Auto	RM	Open	Open
RR "B" Hyd. Contl.	1B33F356D	M-69	O.C.	57	Globe SO	Auto	RM	Open	Open
Drywell Inst. Sup.	1IN184	M-70	O.C.	56	Check	Rev. Flow	--	Open	Open
Drywell Inst. Sup.	1IN061	M-70	O.C.	56	Globe MO	Auto	RM	Open	Open
Service Air	1SA081	M-71	I.C.	56	Globe	Man.	--	L. Shut	L. Shut
Service Air	1SA079	M-71	O.C.	56	Globe	Man.	--	L. Shut	L. Shut
HPCS Relief Valve	1E22F014	M-72	O.C.	56	Relief	Over Press.	--	Shut	Shut
RHR Relief Valve	1E12F025A	M-72	O.C.	56	Relief	Over Press.	--	Shut	Shut
LPCS Relief	1E21F018	M-73	O.C.	56	Relief	Over Press.	--	Shut	Shut
RHR Relief Vent	1E12F005	M-73	O.C.	56	Relief	Over Press.	--	Shut	Shut
RHR Relief Vent	1E12F349A	M-73	O.C.	56	Relief	Over Press.	--	Shut	Shut
RHR Vent - HX-A	1E12F073A	M-73	O.C.	56	Globe MO	RM	Man.	Shut	Shut
RHR Vent - HX-B	1E12F036	M-73	O.C.	56	Relief	Over Press.	--	Shut	Shut
LPCS Relief Vent	1E21F031	M-73	O.C.	56	Relief	Over Press.	--	Shut	Shut
RHR Relief	1E12F403	M-73			Relief	Over Press.	--	Shut	Shut
PC Chill Wtr. "B"	1VP006B	M-74	O.C.	56	Gate MO	Auto	RM	Open	Open
PC Chill Wtr. "B"	1VP068O	M-74	I.C.	56	Check	Rev. Flow	--	Open	Open
PC Chill Wtr. "B"	1VP012B	M-75	O.C.	56	Gate MO	Auto	RM	Open	Open
PC Chill Wtr. "B"	1VP045B	M-75	I.C.	56	Gate MO	Auto	RM	Open	Open
PC Chill Wtr. "A"	1VP012A	M-76	O.C.	56	Gate MO	Auto	RM	Open	Open
PC Chill Wtr. "A"	1VP045A	M-76	I.C.	56	Gate MO	Auto	RM	Open	Open

VE POSITION: POST LOCA	VE POSITION: POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
t	--	L,B	3/4	STD.	Yes	B	1	N	No	See Note 28	
t	--	L,B	1/2	STD.	Yes	B	1	N	No	See Note 28	76
t	--	L,B	1/2	STD.	Yes	B	1	N	No	See Note 28	
t	--	L,B	3/4	STD.	Yes	B	1	N	No	See Note 28	
t	--		3	STD.	No	B	1	Y	No	See Note 25	82
t	As is	B,L,M,Z	3	STD.	Yes	B	1	Y	No	See Note 25	76
Shut	As is	--	2 1/2	--	No	B	1	Y	No		
Shut	As is	--	2 1/2	--	No	B	1	Y	No		
t	--	--	1	--	No	B	1	Y	No	Set @ 100 psig See note 18	73
t	--	--	2	--	No				No	Set @ 500 psig See note 18	82
t	--	--	3	--	No	B	1	Y	No	Set @ 500 psig See note 18	
t	--	--	1 1/2	--	No	B	1	Y	No	Set @ 220 psig See note 18	
t	--	--	1 1/2	--	No	B	1	Y	No	Set @500 psig See note18	79
t	As is	--	1 1/2	STD.	No	B	1	Y	No	See Note 18	
t	--	--	4	--	No	B	1	Y	No	Set @ 100 psig See note 18	79
t	--	--	2	--	No	B	1	Y	No	Set @ 100 psig See note 18	
t	--	--									73
t	As is	B,L	8	STD.	No	B	1	Y	No		
t	--	--	8	--	No	B	1	Y	No		82
t	As is	B,L	8	STD.	No	B	1	Y	No		
t	As is	--	8	STD.	No	B	1	Y	No	See Note 18	82
t	As is	B,L	8	STD.	No	B	1	Y	No		
t	As is	B,L	8	STD.	No	B	1	Y	No	See Note 18	82

TABLE 6

ISOLATION VALVE SUMMARY FOR LIN

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION				V
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN	
PC Chill Wtr. "A"	1VP006A	M-77	O.C.	56	Gate MO	Auto	RM	Shut	Shut	
PC Chill Wtr. "A"	1VP068A	M-77	I.C.	56	Check	Rev. Flow	--	Shut	Shut	
Through DW Floor	--	M-78	--	--	--	--	--	--	--	
Through DW Floor	--	M-79	--	--	--	--	--	--	--	
Through DW Floor	--	M-80	--	--	--	--	--	--	--	
Through DW Floor	--	M-81	--	--	--	--	--	--	--	
Through DW Floor	--	M-82	--	--	--	--	--	--	--	
Through DW Floor	--	M-83	--	--	--	--	--	--	--	
Through DW Floor	--	M-84	--	--	--	--	--	--	--	
Through DW Floor	--	M-85	--	--	--	--	--	--	--	
Through DW Floor	--	M-86	--	--	--	--	--	--	--	
Through DW Floor	--	M-87	--	--	--	--	--	--	--	
Through DW Floor	--	M-88	--	--	--	--	--	--	--	
Through DW Floor	--	M-89	--	--	--	--	--	--	--	
Through DW Floor	--	M-90	--	--	--	--	--	--	--	
Through DW Floor	--	M-91	--	--	--	--	--	--	--	
Through DW Floor	--	M-92	--	--	--	--	--	--	--	

VALVE POSITION		CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS
POST LOCA	POWER FAILURE									
Shut	As is	B,L	8	STD.	No	B	1	Y	No	
Shut	--	--	--	--	No	B	1	Y	No	82
--	--	--	--	--	--	--	--	--	--	
--	--	--	--	--	--	--	--	--	--	
--	--	--	--	--	--	--	--	--	--	
--	--	--	--	--	--	--	--	--	--	
--	--	--	--	--	--	--	--	--	--	
--	--	--	--	--	--	--	--	--	--	
--	--	--	--	--	--	--	--	--	--	73
--	--	--	--	--	--	--	--	--	--

TABLE 6

ISOLATION VALVE SUMMARY FOR LIN

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION				V
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN	
In Supply	1IN012	M-93	1.C.	56	Globe AO	Auto	RM	Open	Open	
In Supply	1IN013	M-93	O.C.	56	Globe AO	Auto	RM	Open	Open	
Spare	--	M-94	--	--	--	--	--	--	--	
ESS Inst. Air A	1IN008A	M-95	O.C.	57	Globe MO	RM	Man.	Open	Open	
Spare	--	M-96	--	--	--	--	--	--	--	
RHR Relief Vent	1E12F348B	M-97	O.C.	56	Relief	Overpress	--	Shut	Shut	
RHR Relief Vent	1E12F030	M-97	O.C.	56	Relief	Overpress	--	Shut	Shut	
RHR Relief Vent	1E12F025B	M-97	O.C.	56	Relief	Overpress	--	Shut	Shut	
RHR Relief Vent	1E12F025C	M-97	O.C.	56	Relief	Overpress	--	Shut	Shut	
RHR Hx "B" Vent	1E12F73B	M-97	O.C.	56	Globe MO	RM	Man.	Shut	Shut	
RHR "A"Test	1E12F024A	M-98	O.C.	56	Globe MO	Auto	RM	Shut	Shut	
Spare	--	M-99	--	--	--	--	--	--	--	
Spare	--	M-100	--	--	--	--	--	--	--	
Drywell Purge Inlet	1VQ001A	M-101	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	
Drywell Purge Inlet	1VQ001B	M-101	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	
Drywell Purge Exh	1VQ002A	M-102	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	
Drywell Purge Exh	1VQ002B	M-102	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	
Drywell Purge &Rlf	1VQ005A	M-102	O.C.	56	Globe SO	RM	Mar.	Shut	Shut	
Drywell Purge &Rlf	1VQ005B	M-102	O.C.	56	Globe SO	RM	Man.	Shut	Shut	

VALVE POSITION		CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
POST LOCA	POWER FAILURE										
hut	F.C.	B,L,M,Z	2.5	STD	Yes	B	1	Y	No		
hut	F.C.	B,L,M,Z	2.5	STD	Yes	B	1	Y	No		76
--	--	--	--	--	--	-	-	-	--	Type "A" Test	
pen-Shut	As Is	--	1	STD	Yes	B	1	Y	Yes		
--	--	--	--	--	--	-	-	-	--		
hut	--	--	1.5	--	No	B	1	Y	No	Set @ 500 psig See note 18	
hut	--	--	2	--	No	B	1	Y	No	Set @ 150 psig See note 18	
hut	--	--	2	--	No	B	1	Y	No	Set @ 500 psig See note 18	82
hut	--	--	1.5	--	No	B	1	Y	No	Set @ 500 psig See note 18	
hut	As Is	--	1.5	STD	No	B	1	Y	No	See Note 18	
											73
hut	As Is	A,L	10	STD	Yes	B	1	Y	No	See Note 18	76 82
--	--	--	--	--	--	-	-	-	--	Type "A" Test	
--	--	--	--	--	--	-	-	-	--	Type "A" Test	
hut	Shut	B,L,M,Z	18	5	Yes	B	1	Y	No	See Note 11	
hut	Shut	B,L,M.Z	18	5	Yes	B	1	Y	No	See Notes 11, 18	76 82
hut	Shut	B,L,M.Z	18	5	Yes	B	1	Y	No	See Notes 11, 18	
hut	Shut	B,L,M,Z	18	5	Yes	B	1	Y	No	See Note 11	
hut	Shut	B,L,M,Z	2	STD	Yes	B	1	Y	No	See Notes 11,	76
hut	Shut	B,L,M,Z	2	STD	Yes	B	1	Y	No	See Note 11	

TABLE 6.

ISOLATION VALVE SUMMARY FOR LINE

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION				VA
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN	
Wetwell Purge Exh	1VQ004A	M-103	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	Sh
Wetwell Purge Exh	1VQ004B	M-103	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	Sh
Comb. Gas Control	1T49F004A	M-103	O.C.	56	Globe MO	RM	Man.	Shut	Shut	Op
Comb. Gas Control	1T49F004B	M-103	O.C.	56	Globe MO	RM	Man.	Shut	Shut	Op
Wetwell Purge Inlt	1VQ003A	M-104	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	Sh
Wetwell Purge Inlt	1VQ003B	M-104	O.C.	56	Butterfly AO	Auto	RM	Shut	Open	Sh
Purge Air Supply	1VQ008A	M-104	O.C.	56	Globe SO	Auto	RM	Shut	Shut	Sh
Purge Air Supply	1VQ008B	M-104	O.C.	56	Globe SO	Auto	RM	Shut	Shut	Sh
RCIC Vacuum Bkr	1E51F086	M-104	O.C.	56	Globe MO	Auto	RM	Open	Open	Op
RHR Relief Vent	1E12F055B	M-105	O.C.	56	Relief	Overpress	--	Shut	Shut	Sh
RHR Relief Vent	1E12F055A	M-106	O.C.	56	Relief	Overpress	--	Shut	Shut	Sh
Vacuum Bkr.	--	M-107	--	--	--	--	--	--	--	
Vacuum Bkr.	--	M-108	--	--	--	--	--	--	--	
Vacuum Bkr.	--	M-109	--	--	--	--	--	--	--	
Vacuum Bkr.	--	M-110	--	--	--	--	--	--	--	

VE POSITION: POST LOCA	VE POSITION: POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
	Shut	B,L,M,Z	18	5	Yes	B	1	Y	No	See Notes 11, 18	
	Shut	B,L,M,Z	18	5	Yes	B	1	Y	No	See Note 11	82
Shut	As Is	--	2	STD	Yes	B	1	Y	No	2 Valves in Parallel	76
Shut	As Is	--	2	STD.	Yes	B	1	Y	No	2 Valves in Parallel	
	Shut	B,L,M,Z	18	5	Yes	B	1	Y	No	See Note 11	82
	Shut	B,L,M,Z	18	5	Yes	B	1	Y	No	See Notes 11, 18	
	Shut	B,L,M,Z	2	STD	Yes	B	1	Y	No	See Notes 11,	76
	Shut	B,L,M,Z	2	STD	Yes	B	1	Y	No	See Note 11	
Shut	As Is	L,V4	2	STD	Yes	B	1	Y	No		
	--	--	8	--	No	B	1	Y	No	Set @ 500 psig See note 18	73
	Shut	--	8	--	No	B	1	Y	No	Set @ 500 psig See note 18	79
	--	--	--	--	--	-	-	-	--	Through Drywell Floor	
	--	--	--	--	--	-	-	-	--	Through Drywell Floor	
	--	--	--	--	--	-	-	-	--	Through Drywell Floor	
	--	--	--	--	--	-	-	-	--	Through Drywell Floor	

TABLE 6

ISOLATION VALVE SUMMARY FOR LIN

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION				V
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN	
Jet Pump Flow	1B21F374	I-1A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F373	1B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
RWCU Flow	1G33F316	1C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Spare		1D								
Jet Pump Flow	1B21F372	1E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F371	1F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F370	I-2A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F369	2B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Pres Above Core Pl	1B21F375	2C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Spare		2D								
Jet Pump Flow	1B21F368	2E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F367	2F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F366	I-3A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F365	3B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Pres Below Core Pl	1B21F377	3C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Spare		3D								
Jet Pump Flow	1B21F364	3E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F363	3F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F389	I-4A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F388	4B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Main Steam Flow A	1B21F472A	4C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Main Steam Flow A	1B21F473A	4D	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F387	4E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F386	4F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F385	I-5A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Jet Pump Flow	1B21F384	5B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open	O
Spare		5C								
Recirc Seal Purge	1B33F017A 1B33F013A	5D	O.C./ O.C.	55	Check/Check	Rev. Flow/ Rev. Flow	-	Open	Shut/ Open	O S

VE POSITION: POST LOCA	VE POSITION: POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	Yes	See Note 22	76
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	73
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	Yes	See Note 22	76
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	Yes	See Note 22	76
n	As Is	-	3/4	-	No	B	1	N	Yes	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n	As Is	-	3/4	-	No	B	1	N	No	See Note 22	
n/	--	-	3/4	-	No	B	1	Yes	No		82

TABLE 6

ISOLATION VALVE SUMMARY FOR LI

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION			
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN
Primary Containment Air Sample	1CM176	I-19A	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM177	19A	O.C.	56	Ball	Manual	-	Shut	Shut
Air Sample Control Rod Drive Pedestal	1CM194	19B	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM195	19B	O.C.	56	Ball	Manual	-	Shut	Shut
Air Samp Jet Pump Valve	1CM196	19C	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM197	19C	O.C.	56	Ball	Manual	-	Shut	Shut
Air Samp Jet Pump Valve	1CM198	19D	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM199	19D	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		19E							
Rx Recirc Sample	1B33F019	19F	I.C.	55	Globe	Auto	Remote	Open	Shut
	1B33F020	19F	O.C.	55	Globe	Auto	Manual	Open	Shut
Spare		19G							
Suppression Chamber Water Level	1CM039	I-20	O.C.	56	Excess Flow	Rev. Flow	-	Open	Open
RCIC System Steam Leak Detection 1A	1E51F318B	I-21A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
RCIC System Steam Leak Detection 1A	1E51F317B	21B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Recirc Pump 1B Seal Purge Water	1B33F017B	21C	I.C.	55	Check	Rev. Flow	-	Open	Shut/ Open
	1B33F013B	21C	O.C.	55	Check	Rev. Flow	-	Open	Shut/ Open
Spare		21D							
RCIC System Steam Leak Detection 2A	1E51F318A	21E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
RCIC System Steam Leak Detection 2A	1E51F317A	21F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Spare		I-22A							
Spare		22B							
Spare		22C							
Spare		22D							
Spare		22E							
Spare		22F							
Spare		I-23A							
Spare		23B							

VALVE POSITION											
POST LOCA	POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11 & 18	
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11 & 18	
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11 & 18	
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11 & 18	
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
Shut	Shut	B, C	3/4	STD	No	B	1	Yes	No		73
Shut	Shut	B, C	3/4	STD	No	B	1	Yes	No		
Open	-	-	3/4	-	Yes	B	1	N	Yes	See Note 24	
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	76
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
hut/ Open	-	-	3/4	-	No	B	1	Y	No		
hut/ Open	-	-	3/4	-	No	B	1	Y	No		82
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	76
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	

TABLE

ISOLATION VALVE SUMMARY FOR L

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION			
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTD
Main Steam Flow A	1B21F468A	I-40A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow A	1B21F469A	40B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow B	1B21F472B	40C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow B	1B21F473B	40D	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow C	1B21F468C	40E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow C	1B21F469C	40F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Spare		I-41							
Spare		I-42							
Spare		I-43							
Spare		I-44							
Containment Flood Level (R.T.)	1CM033A	I-45	O.C.	56	Excess Flow	Rev. Flow	-	Open	Open
Containment Flood Level (R.T.)	1CM033B	I-46	O.C.	56	Excess Flow	Rev. Flow	-	Open	Open
Reactor Recirc Pump 1B - Flow	1B33F373	I-47A	O.C.	55	Globe	Manual	-	Shut	Shut
	1B33F374	47A	O.C.	55	Globe	Manual	-	Shut	Shut
Spare		47B							
Spare		47C							
Reactor Recirc Pump 1A - Flow	1B33F367	47D	O.C.	55	Globe	Manual	-	Shut	Shut
	1B33F368	47D	O.C.	55	Globe	Manual	-	Shut	Shut
Spare		I-48A							
Spare		48B							
Reactor Water Cleanup Inlet	1G33F315	48C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Cleanup Inlet	1G33F314	48D	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Spare		48E							
Spare		48F							

VALVE POSITION		CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
POST LOCA	POWER FAILURE										
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	76
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
											73
Open	-	-	3/4	-	No	B	1	N	No	See Note 24	
Open	-	-	3/4	-	No	B	1	N	No	See Note 24	
Shut	Shut	-	3/4	-	No	B	1	N	No		
Shut	Shut	-	3/4	-	No	B	1	N	No		82
Shut	Shut	-	3/4	-	No	B	1	N	No		
Shut	Shut	-	3/4	-	No	B	1	N	No		82
Open	-	-	3/4	-	No	B	1	N	No	See Note 22	
Open	-	-	3/4	-	No	B	1	N	No	See Note 22	

TABLE

ISOLATION VALVE SUMMARY FOR LI

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION			
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDO
Spare		I-49A							
Spare		49B							
Spare		49C							
Reactor Recirc Pump 1A - Flow	1B33F369	49D	O.C.	55	Globe	Manual	-	Shut	Shut
	1B33F370	49D	O.C.	55	Globe	Manual	-	Shut	Shut
Reactor Recirc Pump 1B - Flow	1B33F371	49E	O.C.	55	Globe	Manual	-	Shut	Shut
	1B33F372	49E	O.C.	55	Globe	Manual	-	Shut	Shut
Spare		49F							
Air Sample Reactor Head Closure	1CM200	I-50A	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM201	50A	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		50B							
Spare		50C							
Air Sample Reactor Head Closure	1CM202	50D	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM203	50D	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		50E							
Spare		50F							
Suppression Chamber Water Level	1CM041	I-51	O.C.	56	Excess Flow	Rev. Flow	-	Open	Open
Contain Flooding Level (P.I.)	1CM032A	I-52	O.C.	56	Excess Flow	Rev. Flow	-	Open	Open
Contain Flooding Level (P.I.)	1CM032B	I-53	O.C.	56	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow B	1B21F470B	I-54A	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow B	1B21F471B	54B	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow C	1B21F474C	54C	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow C	1B21F475C	54D	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow D	1B21F474D	54E	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Main Steam Flow D	1B21F475D	54F	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
		I-55 to I-63							

LVE POSITION											
POST LOCA	POWER FAILURE	CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
hut	-	-	3/4	-	No	B	1	N	No		
hut	-	-	3/4	-	No	B	1	N	No		82
hut	-	-	3/4	-	No	B	1	N	No		
hut	-	-	3/4	-	No	B	1	N	No		
hut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
hut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
hut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	73
hut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
pen	-	-	3/4	-	Yes	B	1	N	No	See Note 23	
pen	-	-	3/4	-	No	B	1	N	No	See Note 24	
pen	-	-	3/4	-	No	B	1	N	No	See Note 24	
pen	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
pen	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
pen	-	-	3/4	-	No	B	1	N	Yes	See Note 22	76
pen	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
pen	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
pen	-	-	3/4	-	No	B	1	N	Yes	See Note 22	
										Through Drywell Floor	

TABLE

ISOLATION VALVE SUMMARY FOR LI

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION			
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDOWN
Drywell Atom Sample	1CM006	I-65A	O.C.	56	Globe	Auto	RM	Open	Open
	1CM005	65A	O.C.	56	Globe	Auto	RM	Open	Open
Post LOCA	1CM007	65A	O.C.	56	Globe	Auto	RM	Shut	Shut
Post LOCA	1CM008	65A	O.C.	56	Globe	Auto	RM	Shut	Shut
Spare		65B							
Near Safety/Relief Valves	1CM204	65C	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM205	65C	O.C.	56	Ball	Manual	-	Shut	Shut
Near Safety/Relief Valves	1CM206	65D	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM207	65D	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		65E							
Spare		65F							
Air Sample MS ISO V 1B21F022A&B	1CM178	I-66A	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM179	66A	O.C.	56	Ball	Manual	-	Shut	Shut
Air Sample MS ISO V 1B21F021C&D	1CM180	66B	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM181	66B	O.C.	56	Ball	Manual	-	Shut	Shut
Air Sample Recirc Pump B In Valve	1CM184	66C	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM185	66C	O.C.	56	Ball	Manual	-	Shut	Shut
Air Sample Recirc Pump B Out Valve	1CM188	66D	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM189	66D	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		66E							
Air Sample Recirc Pump B Seals	1CM192	66F	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM193	66F	O.C.	56	Ball	Manual	-	Shut	Shut
Air Sample Recirc Pump A In Valve	1CM182	I-67A	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM183	67A	O.C.	56	Ball	Manual	-	Shut	Shut
Air Sample Recirc Pump A Out Valve	1CM186	67B	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM187	67B	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		67C							
Air Sample Recirc Pump A Seals	1CM190	67D	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM191	67D	O.C.	56	Ball	Manual	-	Shut	Shut
Spare		67E							

VE POSITION		CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.E.4.2	REMARKS	
POST LOCA	POWER FAILURE										
ut	Shut	B,L,M,Z	1/2	STD	Yes	B	1	Y	Yes		
ut	Shut	B,L,M,Z	1/2	STD	Yes	B	1	Y	Yes		
en	Shut	B,L,M,Z (Open)	3/4	STD	Yes	B	1	Y	Yes		
en	Shut	B,L,M,Z (Open)	3/4	STD	Yes	B	1	Y	Yes		
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11, 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	73 82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82
ut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18	
ut	-	-	1/2	-	No	B	1	Y	No	See Note 11	82

TABLE

ISOLATION VALVE SUMMARY FOR LI

SYSTEM NAME	VALVE NUMBER	PRIMARY CONT. PENT. NUMBER	VALVE LOCATION	NRC DESIGN CRITERIA	VALVE TYPE	ACTUATION			
						PRIMARY MODE	SECONDARY MODE	NORMAL	SHUTDO
Air Sample RHR Pump Supply	1CM208	I-67F	O.C.	56	Ball	Manual	-	Shut	Shut
	1CM209	67F	O.C.	56	Ball	Manual	-	Shut	Shut
Reactor Water Level & Pressure	1B21F353	I-68	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level	1B21F355	I-69	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level	1B21F354	I-70	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level & Pressure	1B21F357	I-71	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level	1B21F358	I-72	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level & Pressure	1B21F359	I-73	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level & Pressure	1B21F361	I-74	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level	1B21F360	I-75	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level & Pressure	1B21F351	I-76	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
Reactor Water Level	1B21F352	I-77	O.C.	55	Excess Flow	Rev. Flow	-	Open	Open
		I-78 to I-104							

VALVE POSITION		CONTAINMENT ISOLATION SIGNAL	LINE SIZE (INCHES)	CLOSURE TIME (SECONDS)	ENGINEERED SAFETY FEATURE	QUALITY GROUP	SEISMIC CLASS	TYPE "C" TEST	ESSENTIAL per II.2.4.2	REMARKS
POST LOCA	POWER FAILURE									
Shut	-	-	1/2	-	No	B	1	Y	No	See Notes 11 and 18
Shut	-	-	1/2	-	No	B	1	Y	No	See Note 11
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
Open	-	-	3/4	-	No	B	1	N	Yes	See Note 22
										Through drywell floor

82 76 73

TABLE 6.2-8 (Cont'd)

Th. flowout of the CRD is restricted through the HCU by performance test requirements to ensure that HCU leakage does not exceed 0.2 gpm. The maximum leakage expected for these penetrations is 0.2 gpm per HCU. If a single failure is assumed, the maximum leakage would be 3 gpm. Seismic tests have demonstrated the seal integrity of the CRD system. Maximum leakage following these tests did not exceed 3 gpm.

The system design criteria are as follows:

Component	Seismic Category	Quality Group Classification	Quality Assurance Classification
Valves; return line within containment boundary	I	A	I
Valves; insert and withdraw	I	B	I
Return line piping within isolation valves	I	A	I
Insert and withdraw line piping	I	B	I

43

The CRD insert and withdraw lines are compatible with the criteria intended by 10 CFR 50, Appendix J, for Type C testing, since the acceptance criterion for Type C testing allows demonstration of fluid leakage rates by associated bases. The maximum leakage expected has been factored in with the total allowable containment penetration leakage and determined to be acceptable.

10. Main steamline isolation valves to be type "C" tested in accordance with Subsection 5.5.4. Leakage is exempt from Appendix J total measured leakage criteria. | 31

11. Test pressure will be applied between the valves. The total leakage recorded will be assigned to each penetration.

12. Penetration paths for M-18 and M-24 will be tested si ultaneously. | 14 | 82

13. Deleted.

| 43

14. Deleted.

15. Deleted. | 38

68

7.6.1.3.4 Process Liquid Radiation Monitor Subsystem

7.6.1.3.4.1 System Identification

The purpose of the process liquid radiation monitors is to indicate to operations personnel when the radioactivity level of a liquid stream being released to the environs exceeds preestablished limits. The purpose is also to indicate when the radioactivity level in a stream not being released to the environs exceeds preestablished limits.

7.6.1.3.4.2 Power Sources

The 125 Vdc Buses A and B are the power sources for this subsystem.

7.6.1.3.4.3 Equipment Design

Circuit Description

The process liquid radiation monitor is shown in Figure 7.6-1. Specifications are given in Table 7.6-2. Three in-line channels plus four offline samplers are provided. One offline channel monitors the service water from the turbine building concentrator condensers, two in-line channels monitor the reactor building closed cooling water system, the third in-line channel monitors the discharge from the liquid radwaste system, two offline sampler channels monitor the service water discharge from each RHR heat exchanger, and the offline sampler monitors the service water discharge downstream of the radwaste addition. 82

Each channel has a scintillation detector and a radiation monitor. The detector monitors the process liquid for gamma activity as the liquid flows through either the process pipe (in the case of the in-line monitoring channels), or a control volume (in the case of the offline samples). The monitors and service water discharge recorders are located in the control room. The monitors are

TABLE 7.6-9

LOCATIONS FOR AREA RADIATION MONITORING SENSORS*

INSTRUMENT SYMBOL	RADIATION ZONE AND IDENTIFICATION NUMBER	RADIATION DETECTOR NO.	DESCRIPTION	PLAN ELEVATION	FIGURE NUMBER	COORD.	RANGE (mr/hr)	ALARM SETPOINT (mr/hr)	LOCAL ALARM	CCS SIGN	
ARM-12	ARM-12	1D21-R002L	Reactor Building Control Rod Drive Hydraulic Units	546'-0"	12.1-1	F-6	0.1-1000	5	Yes	19-E	
ARM-13	ARM-13	1D21-R002M	Reactor Building Southwest Corner	546'-0"	12.1-1	E-7	0.1-1000	5	No	21-F	
ARM-20	ARM-20	1D21-R002A	Main Control Room	546'-0"	12.1-1	D-7	0.01-100	1	No	21-E	
ARM-25	ARM-25	1D21-R002Y	Turbine Building East Corridor	546'-0"	12.1-1	D-5	0.1-1000	5	No	13-F	
ARM-26	ARM-26	1D21-R002AR	Turbine Building Electro Hydraulic Control Area	546'-0"	12.1-1	C-5	0.1-1000	15	No	15-E	
ARM-27	ARM-27	1D21-R002AR	Turbine Building Entrance	546'-0"	12.1-1	B-5	0.1-1000	30	No	12-E	
ARM-14	ARM-14	1D21-R002Y	Reactor Building Clg Drive Area	525'-3"	12.1-2	E-7	0.1-1000	5	Yes	21-E	
ARM-15	ARM-15	1D21-R002S	Reactor Building Dry Well Equipment Hatch	525'-3"	12.1-2	F-6	0.1-1000	5	No	19-E	
ARM-13	ARM-13	1D21-R002A	Reactor Building Tie Cubicle	525'-3"	12.1-2	E-7	1.0-10^4	100	Yes	21-E	
ARM-21	ARM-21	1D21-R002Y	Switchgear Area	525'-3"	12.1-2	D-6	0.1-1000	5	No	16-E	
ARM-28	ARM-28	1D21-R002AC	Air Ejector Area	525'-3"	12.1-2	C-7	0.1-1000	5	No	20-E	
ARM-29	ARM-29	1D21-R002AD	Turbine Building East Corridor	521'-0"	12.1-2	C-6	0.1-1000	5	No	17-E	
ARM-30	ARM-30	1D21-R002AE	Condensate Demineralizer Area	521'-0"	12.1-2	B-7	0.1-1000	5	No	24-F	81
ARM-38	ARM-38	1D21-R002AM	Dry Active Waste Storage Area	521'-0"	12.1-2	A-7	0.1-1000	5	No	28-E	
ARM-39	ARM-39	1D21-R002AN	Solid Radwaste Area	521'-0"	12.1-2	A-6	0.1-1000	5	No	22-E	
ARM-31	ARM-31	1D21-R002A	Roots Regenerator Cubicle	521'-0"	12.1-2	B-6	1.0-10^4	100	Yes	19-F	
ARM-41	ARM-41	1D21-R002B	Decontamination Room	521'-0"	12.1-3	F-4	0.1-100	1	No	9-B	
ARM-22	ARM-22	1D21-R002Y	Switchgear Area	519'-6"	12.1-3	D-6	0.1-1000	5	No	19-E	
ARM-42	ARM-42	1D21-R002AN	Radwaste Building Pump Area	511'-3"	12.1-4	A-7	0.1-1000	5	No	27-E	
ARM-16	ARM-16	1D21-R002P	Reactor Building Hvac Vent (Northwest)	503'-6"	12.1-4	E-7	0.1-1000	5	No	26-E	
ARM-17	ARM-17	1D21-R002B	Reactor Building Southeast Corner	503'-6"	12.1-4	F-6	0.1-1000	5	No	19-E	74
ARM-23	ARM-23	1D21-R002W	High Level Laboratory	496'-0"	12.1-4	D-6	0.1-1000	5	No	17-G	82
ARM-32	ARM-32	1D21-R002AF	Turbine Building Southeast Corner	496'-0"	12.1-4	C-6	0.1-1000	5	No	26-E	
ARM-33	ARM-33	1D21-R002AG	Make-Up Demineralizer Room	496'-0"	12.1-4	B-6	0.1-1000	5	No	22-F	
ARM-40	ARM-40	1D21-R002AP	Radwaste Area	496'-0"	12.1-4	A-7	0.1-1000	5	No	24-F	
ARM-18	ARM-18	1D21-R002S	Reactor Building High Pressure Core Spray Equipment Room	475'-6"	12.1-5	F-6	0.1-1000	5	No	14-E	
ARM-19	ARM-19	1D21-R002T	Reactor Building Suppression Pool Area	475'-6"	12.1-5	E-6	0.1-1000	5	Yes	15-E	
ARM-24	ARM-24	1D21-R002X	Offgas Treatment System Area	471'-5"	12.1-5	D-7	0.1-1000	5	No	26-E	
ARM-34	ARM-34	1D21-R002AH	Discharge Tanks Area	471'-5"	12.1-5	C-7	0.1-1000	5	No	26-F	
ARM-45	ARM-45	1D21-R002AS	Turbine Building Hear Panels	471'-5"	12.1-5	C-6	0.1-1000	5	No	17-E	81
ARM-36	ARM-36	1D21-R002AK	Condensate Booster Pumps Area	471'-5"	12.1-5	B-6	0.1-1000	5	No	16-E	
ARM-35	ARM-35	1D21-R002AJ	Turbine Building Waste Pumps Area	471'-5"	12.1-5	B-6	0.1-1000	5	No	16-E	
ARM-42	ARM-33	1D21-R002/C	Hear Waste Sample Tank Pumps	471'-5"	12.1-5	B-7	1.0-10^4	100	Yes	26-E	
ARM-2	ARM-34	1D21-R002A	Reactor Building Refueling Area	527'-6"	12.1-6	E-7	0.1-1000	10	Yes	24-D	81
ARM-3	ARM-35	1D21-R002B	Reactor Building Equipment Hatch	627'-6"	12.1-6	E-7	0.1-1000	5	Yes	28-E	
ARM-A	ARM-36	1D21-R002C	Reactor Building Northwest Corner	627'-6"	12.1-6	D-7	0.1-1000	5	Yes	28-D	
ARM-5	ARM-37	1D21-R002D	Reactor Building Equipment Storage Pool	627'-6"	12.1-6	C-6	0.1-1000	5	Yes	15-C	
ARM-1	ARM-38	1D21-R004	Reactor Building Refueling Floor High Level	627'-6"	12.1-6	C-6	10^2-10^5	1000	Yes	23-F	81
ARM-6	ARM-39	1D2-R002E	Reactor Building New Fuel Storage Vault	581'-6"	12.1-6	E-5	0.1-1000	15	Yes	35-E	
ARM-7	ARM-40	1D21-R002F	Reactor Building Fuel Pool Filters	595'-6"	12.1-6	E-4	0.1-1000	5	No	18-A	
ARM-8	ARM-41	1D21-R002G	Reactor Building Reactor Water Cleanup Area	595'-6"	12.1-6	D-4	0.1-1000	5	No	15-F	
ARM-10	ARM-42	1D21-R002J	Reactor Building Standby Liquid Control Area	570'-6"	12.1-6	C-1	0.1-1000	5	No	19-E	
ARM-37	ARM-43	1D21-R002AL	Radwaste Control Room	527'-0"	12.1-2	B-7	0.1-1000	5	Yes	27-E	
	ARM-44	1RE-CR086	Post Accident High Range In-Containment Detector A	In Penetration N-22 @ El. 529'-0" Az. 283°	17.1-3	E-6	10^1-$10^{7\dagger}$	**	No	18-D	
	ARM-45	1RE-CR087	Post Accident High Range In-Containment Detector B	In Penetration N-57 @ El. 538'-7" Az. 159°	12.1-1	F-6	10^1-$10^{7\dagger}$	**	No	22-C	
	ARM-46	1RE-AR001	Technical Support Center Area Radiation Detector	**	--	--	0.1-1000	5	Yes	**	82
	ARM-47	1RE-AR002	Refueling Floor Low Range Detector	**	**	**	1-10^4	**	**	**	81
	PIN		Portal Monitor	Ground	**	**	160-7000***	background	Yes	**	
	PIN		Portal Monitor	546'-0"	12.1-1	B-7	160-7000***	background	Yes	23-F	
	PIN		Portal Monitor	546'-0"	12.1-1	C-4	160-7000***	background	Yes	C-F	

* Detector locations are shown in figures 12.1-1 through 12.1-7.
** Information to be provided.
*** Range given in counts per minute (cpm) where 1000 cpm = 1 mr/hr.
† Channel 9 is a spare.

8.3.1.11.1 Physical Separation

Each Class 1E component is assigned to one of three Class 1E divisions. Class 1E components with redundant safety functions are assigned to different divisions (see Table 8.3-16). Class 1E components of one division are physically separated from Class 1E components of the other divisions. Class 1E components are physically separated from non-Class 1E high-energy components that could cause loss of redundancy as a result of a design-basis event.

8.3.1.11.1.1 4160-Volt Switchgear Separation

The 4.16-kV switchgear is located in the auxiliary building. Each of the three divisions of switchgear is located in a separate room (elevations 510 feet, 525 feet, and 546 feet). The auxiliary building is a Seismic Category I structure.

8.3.1.11.1.2 480-Volt Unit Substation Separation

12

The 480-volt unit substations which supply power to Class 1E systems are located by division in the same rooms in the auxiliary building as the 4.16-kV switchgear.

8.31.1.11.1.3 Motor Control Center Separation

The 11 motor control centers which serve Class 1E loads are in the auxiliary building in the same room with their associated division switchgear; in the reactor building at elevations 503 ft 6 in. and 525 ft 7 in.; and in the service water building in three separate rooms at elevation 450 ft.

82

8.3.1.11.1.4 120-Volt Panel Separation

The two panels which supply power to Class 1E loads are located in the auxiliary building at elevation 521 feet (in the auxiliary equipment room at the northern end of the building) physically separated from each other by a 3-hour fire wall.

82

8.3.1.11.1.5 Diesel Engine Generator Separation

Each diesel engine generator is located in a separate room in the diesel generator building. 82

8.3.1.11.1.6 D-C Battery and Charger Separation

The 125-Vdc, 250-Vdc, and 24/48-Vdc battery and chargers are located by division in separate rooms in the auxiliary building.

8.3.1.11.2 Cable Tray, Cable Bus, Wireway, and Conduit Separation

ZPS-1 uses cable tray, cable bus, wireway, and conduit to carry and support cable throughout the plant. Examples of these are shown in Figures 8.3-4 and 8.3-5.

8.3.1.11.2.1 The Cable Tray Separation Criteria

The cable tray separation criteria are as follows:

a. Any cable tray that carries a Class 1E engineered safety feature cable is assigned to one of the three Class 1E divisions, denoted by either yellow, blue, or green color identification. 12

   A cable tray that carries only non-Class 1E cables is assigned a white identifcation. The color and segregation codes are shown in Table 8.3-16. 82 16

b. The normal vertical spacing between cable trays of the same Class 1E division is 12 inches measured from the bottom of the upper tray to the top of the lower tray. This is only to facilitate cable installation, but may not always be attainable.

c. Cable trays of different Class 1E divisions are separated by a minimum of 3 feet horizontally measured from the siderail of one tray to the siderail of the adjacent tray.

   Vertical separation is 5 feet measured from the bottom of the top tray to the top of the bottom tray. Where vertical separation cannot be maintained, at 5 feet, metal covers are placed on the bottom tray. Where horizontal separation cannot be maintained, fire barriers are installed between the trays. 82

d. In the cable spreading room, a 1-foot horizontal, 3-foot vertical separation is maintained.

Where this separation cannot be obtained, barriers are utilized. 82

The arrangement of the cable tray system is such that no locally generated missile can damage cabling of more than one division.

In areas with large rotating equipment or high-energy piping, trays at different divisions are separated by a minimum of 20 feet or by a 6-inch wall.

Composite mechanical and electrical drawings were prepared by Sargent & Lundy in developing the tray routings. 82

8.3.1.11.2.2 Cable Bus

Cable bus is used to supply power to the onsite a-c system from the auxiliary transformers to the diesel engine generators, and to the recirculation pumps.

The feeds to the diesel engine generators are Class 1E and, therefore, follow the same separation rules as the cable tray system. 12

8.3.1.11.2.3 Wireways

Wireway (actually a small cable tray) is used to carry the neutron monitoring system from outside the primary containment to the control room.

The wireway system is discussed in Subsection 8.3.1.11.5.

8.3.1.11.2.4 Conduit

Conduit, both rigid and flexible, has the following separation requirements for the three ESF divisions: where conduits of different divisions run in proximity to each other, the minimum separation is 1 inch. In missile areas, the minimum separation is 20 feet. 47

8.3.1.11.3 Control Board and Panel Separation

Class 1E panels containing different Class 1E divisions have a minimum separation between divisions of 6 inches of air space or a floor-to-ceiling fire barrier consisting of steel plate. All terminal blocks, wireway, etc., are maintained 1 inch away from the steel plate barrier. 12 82

Separate Class 1E panels of different divisions are separated by a floor-to-ceiling fire barrier consisting of steel plate. All terminal blocks, wireway, etc., are maintained 1 inch away from the steel plate barrier. 82

Panel ends (ends closed by steel and plates) are considered acceptable fire barriers if blocks, wireways, etc. are 1 inch from the end plate. 12

82

8.3.1.11.4 Containment Electrical Penetrations

Containment electrical penetrations are designed to accommodate the electrical requirements of the drywell equipment. They are grouped functionally according to the type of conductors routed through them. Each penetration seal will have the same basic configuration. The penetrations are flanged to the outside end of the containment with nuts, bolts, washers, and lockwashers. The aperture seal is formed between the penetration flange and the nozzle flange with two concentric silicone O-rings.

The penetration canisters are tubular in form and are slightly longer than the containment nozzle into which they are installed. Electrical conductor feedthroughs are factory installed and sealed to header plates at each end of the canister with metal ferrule compression fittings. This assembly forms a pressure vessel with the electrical conductors passing through and terminating at each end.

This method of arrangement of conductor feedthroughs at the canister ends is based on mating field cable termination requirements. The penetration code for each feedthrough is stamped on the header plate port.

The field terminations are made by splicing, spade connectors, terminal strips, or triaxial connectors, depending on the application of the conductor. 31

The aperture seal is isolated from the canister pressure by an integrator valve (Figure 8.3-8a) prior to factory shipment. Penetration canisters are shipped to the site pressurized. At installation to its containment flange, the canister is depressurized, the pressure gauge and valve assembly is relocated to the edge of the canister flange, and the unit is pressurized with the integrator valve open thus providing continuous leak testing of the canister to flange seal arrangement.

The canister is shipped with an external radiation shield to be installed on the inboard side of the containment in addition to the shielding installed within the canister assembly.

The penetrations are tested "as installed" prior to field terminating cables to verify insulation integrity for each conductor feedthrough and to every other feedthrough in each individual penetration.

There are nineteen electrical penetrations which provide electrical power to equipment inside the primary containment. Cables passing through these penetrations are grouped so that failure of cables in a single penetration cannot prevent safe shutdown. 82

There are two penetrations which supply 69-kV power to the recirculation pump motors.

Two penetrations supply power to motor-operated valves, heaters, etc. Redundant system cables are routed through different penetrations. 12 2

Two penetrations provide a control function and service to valves and heaters rated 120-Vac and 125-Vdc.

Cables of a designated division are routed only in trays, conduit, etc. of that division.

The color codes and segregation codes are shown in Table 8.3-16. 12

8.3.1.12.2.3 Division-Associated Cables

Each non-Class 1E cable which has any part of its length in a safety division raceway or which connects to a Class 1E power system is a division-associated cable and is not routed in a raceway of another safety division from the Class 1E equipment to and including an isolation device. Beyond the isolation device, such a circuit is not subject to these requirements provided it does not again become associated with a Class 1E system. Non-1E cables associated with different safety divisions may be run in a common non-1E raceway. However, once a non-1E cable which is division-associated has been routed in a non-1E tray system, it must then continue to be routed in non-1E trays or conduit to its destination. 76 82

The design basis for electrical separation on the Wm. H. Zimmer Nuclear Power Station includes three ESF electrical divisions: yellow, blue, and green. In addition, there is a non-1E white division which includes nonessential cables. As a design basis, there are cases where nonessential cables are placed in trays dedicated to a specific essential division and there are cases where nonessneital cables are connected to a bus to which essential cables are also connected. As noted above, these are called "associated cables" and are called blue/white, green/white, or yellow/white depending on the interfacing divisions. 82 76 76

This design basis, which is different from the current requirements of Regulatory Guide 1.75, was accepted by NRR in the Zimmer Safety Evaluation Report, NUREG-0528 of January 1979, Section 7.1.2(2) which says: 73

> "Institute of Electrical and Electronic Engineers Standard 384 and Regulatory Guide 1.75, 'Physical Independence of Electrical Systems,' did not exist when the construction permit was issued. However, with the exceptions as indicated in Sections 7 and 8 of this report, the Zimmer Station meets the requirements of these latter criteria."

8.3.1.12.3 Design Control

The design procedures include those which assure adequate physical separation between redundant Class 1E components, and assure the proper assignment of cables to raceways.

8.3.1.12.3.1 Physical Separation 12

The segregation codes of trays, conduits, etc., and electrical equipment division assignments are identified on electrical layout drawings.

Comparison and Justification of ZPS-1 Design vs. Regulatory Guide 1.75 73

Regulatory Guide 1.75 describes a method acceptable to the NRC for ensuring physical independence of the circuits and electric equipment

comprising or associated with Class 1E power system, protection systems, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection systems to perform their safety-related functions. 76

Regulatory Guide 1.75 was issued after the Wm. H. Zimmer Station construction permit. However, the ZPS-1 design meets most of the regulatory positions. Aside from the exceptions noted below, the ZPS-1 design complies with this guide. Position 2 is not applicable.

The ZPS-1 design differs with Position 1 in that the basis for precluding the use of a fault-current-actuated interrupting device as an isolation device within the context of Regulatory Guide 1.75 is questionable. This conclusion seems to preclude the use of a circuit breaker as an isolating device and thus prevents achieving a practical design of an electrical auxiliary system. A qualified (Class 1E) circuit breaker could otherwise fulfill the literal interpretation of Section 3.8 by preventing (by its successful interruption of fault current) a malfunction in one section of a circuit from causing unacceptable influences (I.E., failures) in other sections of the circuit (or other circuits). To preclude Class 1E circuit breakers from these applications casts doubt upon the safety-related portions of the auxiliary power systems in all nuclear plants. Such systems invariably utilize Class 1E metal-clad switchgear and molded case circuit breakers as circuit protection devices.

The Wm. H. Zimmer Station (ZPS-1) design complies with Positions 3, 4, 5, 8, 9, 11, 12, 13, 14, 15, and 16.

With respect to the separation of associated circuits in Position 6 and 7, the ZPS-1 design complies with Section 4.5 of IEEE 384-1974. 73

Although separation of associated circuits in some cases differs from that dictated in Sections 5.1.3 and 5.6.2 of IEEE 384-1974, Class 1E circuits are not degraded below an acceptable level for the following reasons:

a. Cables associated with one Class 1E safety-related division are never routed in cable trays or conduit containing cables of a redundant Class 1E safety-related division. This is true for all general plant areas. Lesser separation than that dictated by IEEE 384-1974 occurs only within control panels located in the control room and auxiliary equipment room and non-1E division trays. 82

b. All cables used to interconnect associated circuits are the same high quality as that utilized in Class 1E circuits, i.e., all associated cables comply with the requirements of IEEE 383-1974. Therefore, this cable has been proven to be highly fire retardant by testing.

c. Cables with separation less than that dictated by Sections 5.1.3 and 5.6.2 of IEEE 384-1974 are constrained to control and instrumentation circuits which, by their very nature,

are low energy circuits. Control circuits are generally 120-Vac or 125-Vac, whereas the insulation rating of the cable utilized at ZPS-1 is 600-V. 76

d. There are no power cables in contact with the control and instrumentation cables in the cable spreading area or in the control room and auxiliary equipment room. Also, there are no high energy sources located within control panels installed in these areas.

e. Fire stops are installed in the bottom entrances of all control panels. 82

Separation of non-Class 1E from Class 1E control and instrumentation circuits is in some cases, less than that required by Sections 5.1.3, 5.1.4, and 5.6.2 of IEEE 384-1974, these circuits have been (or will be) analyzed to show that Class 1E circuits are not degraded below an acceptable level because:

a. Non-Class 1E cables which are not associated cables are routed in separate cable trays from Class 1E cables.

b. Non-Class 1E or associated cables, which are in close proximity (<6 in.) to Class 1E cables at one end are analyzed to ensure that they do not jeopardize cables of a different Class 1E division at their other end. This is being confirmed by a study of installed cables at ZPS-1. 73 82

c. All cables used to interconnect non-Class 1E circuits are the same high quality as that utilized in Class 1E circuits, i.e., all non-Class 1E cables comply with the requirements of IEEE 383-1974. Therefore, this cable has been proven to be highly fire retardant by testing.

d. Cable separation less than that required by Sections 5.1.3, 5.1.4, and 5.6.2 of IEEE 384-1974 is acceptable for control and instrumentation circuits which by their very nature are low energy circuits. Control circuits are generally 120-Vac or 125-Vdc, whereas the insulation rating of the cable utilized at ZPS-1 is 600-V. 82

e. There are no power cables in contact with the control and instrumentation cables in the cable spreading area or in the control room and auxiliary equipment room. Also, there are no high energy sources located within control panels installed in these areas.

f. Fire stops are installed in the bottom entrances of all control panels. 76

in the fuel storage racks results in k_{eff} below 0.90 in a dry condition. In an abnormal condition, if the fuel array were flooded with water and the fuel elements brought to their most reactive spacing, k_{eff} would not exceed 0.95. Information on radiation monitoring of the new fuel storage vaults is provided in Section 7.6.

The new fuel storage racks are designed to meet Seismic Category I requirements. (See Chapter 3.0.) Stresses in a fully loaded rack will not exceed stresses specified by the ASTM standards on lightweight metal alloys when subjected to a 1.5g horizontal acceleration.

The storage rack structure is designed to withstand the impact resulting from a falling weight possessing 2000 ft-lb of kinetic energy. The structural arrangement is such that no lateral displacement of the fuel occurs; therefore, subcritical spacing is maintained.

The new fuel racks are designed to be restrained by holddown bolts to assure that rack spacing does not vary under specified loads. Holddown bolts will restrain the rack in case a stuck fuel assembly is inadvertently hoisted. The rack structure and holddown bolts are designed to maintain the minimum required cell spacing due to forces that might occur in the event that a fuel bundle were to jam in the rack during removal.

The new fuel storage racks are made from aluminum. Materials used for construction are specified in accordance with applicable ASTM specifications (B108-1972, B179-1972, B209-1972, B211-1972, and B221-1972). The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the susceptibility of metals to galvanic corrosion, aluminum and 300-series stainless steel are relatively close together insofar as their coupled potential is concerned. The use of stainless fasteners in aluminum to avoid detrimental galvanic corrosion is a recommended practice and has been used successfully for many years by the aluminum industry. | 82

9.1.1.4 Testing and Inspection

The new fuel storage racks do not require any special testing or inspection for nuclear safety purposes.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

9.1.2.1.1 Safety Design Bases

The fuel array in the fully loaded spent fuel racks is substantially subcritical to prevent fuel barrier damage caused by overheating.

Each spent fuel storage rack containing its full complement of fuel is designed to withstand specified loadings to minimize distortion of the fuel storage arrangement.

The gaseous radwaste system component classification and applicable codes and standards used for design and procurement are provided in Tables 3.2-1 and 3.2-2. All components were purchased after July 1, 1971, as listed in Table 3.2-2. All piping and system components, i.e., vessels, valves, etc. are fabricated from either stainless steel or carbon steel which meet the requirements of the ASTM material standards or ASME Section II 1971 through appropriate addendum in effect at time of purchase. 82 75

The gaseous radwaste system was designed, constructed and tested similar to all other major systems in the plant. Purchase specifications defined all system requirements including quality control; inspection was performed by the manufacturer per the procurement documents and by the constructor during system erection per the design drawings and procedures; all materials were handled, stored, shipped and cleaned in accordance with the established project procedures; and deviations, nonconformance and failures identified for corrective action.

The system is an all welded system except screwed and flanged connections are provided where necessary for maintenance or equipment operation. All process piping is 3/4-inch NPS or greater. All piping 3/4 inch through 2 inch is socket weld. All piping 2-1/2 inch and larger is butt welded using a consumable insert to preclude internal crud traps. All instrument and instrument sample piping is provided with root valves near the main process line to provide shutoff capability. The piping from the process line through the root valve is considered part of the process line. Welding was performed with approved procedures and qualified welders in accordance with ASME Section IX (1971 through Winter 1972 Addendum). All pressurized piping systems are hydrostatically tested to 1.5 times the design pressure. In no case was the test pressure less than 75 psig. The test pressure was held for a minimum time of 10 minutes in accordance with ANSI B31.1.0 (-1967 through Addenda d - 1972) or ASME Section III (1971 through Winter 1972 Addendum) prior to start of visual inspection for leaktightness. The time which the test pressure was maintained was dependent on the extent of the section being hydrostatically tested, but in most cases exceeded 30 minutes. 28 75 73 75 77

11.3.3.1.1.3 Seismic Classification

Components of the off-gas system have been classified and designed as non-Seismic Category I although the charcoal adsorber vessels have in addition been designed to meet the loads and requirements of a Seismic Category I component for conservatism due to uncertainties in system requirements during design and fabrication of the system. To provide additional conservatism in design of the gaseous radwaste system, all system components are located in the auxiliary building, which is a Seismic Category I structure. These measures clearly exceed the seismic criteria prescribed in Regulatory Guide 1.143 (Revision 1). 46 75 77

11.5.2.3 Waste Filter and Floor Drain Filter Sludge

Both the waste filter and floor drain are a pressure pre-coat type, using cellulose fibers (Solka-Floc) as the filtering medium. The filter sludges are pumped to the solid radwaste system from the waste sludge tank and the floor drain sludge tank located at elevation 496 feet in the radwaste building. The solids concentration of the sludge may include as much as 10% iron oxide (particle size range of 0.2 to 5 microns). The filter sludges are expected to have the characteristics listed in Table 11.5-1. 82

11.5.2.4 Concentrator Concentrate Waste

Concentrator concentrate wastes are pumped from the concentrated waste tank located at elevation 527 feet in the radwaste building.

The two types of expected concentrated wastes are described in the following subsections. 82

11.5.2.4.1 Deep Bed Demineralizer Regenerant Solutions

One type of concentrated waste is deep bed demineralizer regenerant solutions, which are approximately 25% (by weight) Na_2SO_4. This is also described in Table 11.5-1.

11.5.2.4.2 Floor Drain Water and Laundry Water Slurries

The second type of concentrated waste is floor drain water and laundry water slurries concentrated 10% to 25% weight total suspended, and dissolved solids, with a pH of 7 to 9.

11.5.3 Equipment and System Description

The solid radwaste system component classification and applicable codes and standards used for design and procurement are provided in Tables 3.2-1 and 3.2-2. All components were purchased after July 1, 1971, as listed in Table 3.2-2. All piping and system components, i.e., vessels, valves, etc. are fabricated from either stainless steel or carbon steel which meet the requirements of the ASTM material standards or ASME Section II (1971 through appropriate addendum in effect at time of purchase). 82 75

The solid radwaste system was designed, constructed and tested in a manner similar to all other major systems in the plant. Purchase specifications defined all system requirements including quality control; inspection was performed by the manufacturer per the procurement documents and by the constructor during system erection per the design drawings and procedures; all materials were handled, stored, shipped and cleaned in accordance with the established project procedures; and deviations, nonconformances and failures identified for corrective action. 82 28

The system is an all welded system except screwed and flanged connections are provided where necessary for maintenance or equipment operation. All process piping is 3/4-inch NPS or greater. All piping 3/4 inch through 2 inch is socket weld. All piping 2-1/2 inch and

c. The system has the capability of being emptied of waste in the solid radwaste metering tanks by either the feed pump or, as a backup, the emergency waste return pump. This feature, in addition to system flushing, will minimize exposure when maintenance is performed.

d. The mixer/feeder is provided with a manual hand crank extended to a low radiation area. This ensures that the small volume of cement paste (less than 1 ft^3) in the mixer feeder unit can be discharged safely, even with the loss of electrical power.

11.5.3.4 Previous Experience

Similar processing systems have been installed and operated at Kewaunee Nuclear Unit #1 (527 MWe PWR, Wisconsin Public Service Corp.); Prairie Island Unit #1 (530 MWe PWR, Northern States Power Co.); Beaver Valley Station Unit #1 (852 MWe PWR, Duquesne Light Co.); Point Beach Unit #1 (497 MWe PWR, Wisconsin Electric Power Co.); and Ft. Calhoun Unit #1 (457 MWe PWR, Omaha Public Power District). In addition, similar systems have been purchased for the Pilgrim Station, the Shoreham Station, the Bellefonte Station, and the Comanche Peak Station. 82

11.5.4 Expected Volumes

The results are presented in Table 11.5-2 which lists for each type of solidified waste the annual weight, volume, and gross activity of the waste processed, the shipping container type, and the annual quantity and shipping weight of the containerized waste.

An isotopic breakdown is referenced for each high-level source in Table 11.5-3. Isotopic inventories for low-level inputs are not listed. The decay realized by onsite storage is the difference between columns 8 and 9 of Table 11.5-3.

11.5.5 Packaging

The initial packaging container for solidified waste at ZPS-1 has a volume of 65 ft^3. During the initial startup of ZPS-1, when the quantity of waste will be high and the activity very low, the system can solidify wastes in 55-gallon drums rather than liners. The system packages the radioactive solid waste in accordance with the applicable regulations, including 49 CFR 170-178. All wastes will be solidified with cement prior to shipment, with the exception of dry compressed waste.

1.5.6 Storage Facilities

The high- and low-level storage areas are shown on Figure 11.5-2, a general arrangement drawing. As shown on Figure 11.5-3, a detail of the storage area, the storage capacities are 18 for high-level liners and 125 for low-level drums. Each liner is the volume equivalent of nine 55-gallon drums. 82

d. erection of physical barriers,

e. posting of warning signs, or

f. administrative control of access.

The station's overall shielding design is such that operating and maintenance procedures are able to take advantage of inherent compartmentalization, remote operability of equipment, and separation of nonradioactive items from areas containing major sources of radiation, where practicable.

Personal monitoring equipment is issued to all station personnel. Daily records are kept of the exposure to individuals employed in radiation areas. Deviations from the allowable exposures specified by the radiation control standards are investigated, and corrective action is taken as needed. Work to be performed in radiation areas is planned to minimize the exposure of the worker to radiation. Temporary shielding, physical barriers, equipment or area decontamination, and/or occupancy duration limits are used to reduce occupational exposures.

Operating data from other boiling water reactor nuclear power plants have been evaluated (References 14 and 16 through 21). These data indicate that operating procedures similar to those proposed herein have been used to limit the average total station personnel exposure.

12.1.6 Estimates of Exposure

Dose rates are estimated during design of the station for various modes of station operation, including full-power operation with anticipated operational occurrences, and for maintenance activities. The nominal design dose rate within the restricted area is less than 1 mrem/hr. Careful management of personnel in the areas where the dose rate exceeds this value (Subsection 12.1.5) ensures that the dose to operating personnel does not exceed the limits specified in 10 CFR 20.

Recent experience, as tabulated in Table 12.1-13, has been evaluated, but it does not account for the more intense precautions being taken to maintain occupational radiation exposure as low as practicable by implementing the guidelines set forth in Regulatory Guide 8.8 (Revision 3). By adhering to these guidelines, it is expected that the overall annual station personnel exposure will be on the order of 300 man-rem. This is consistent with operating experience on a per-man basis as demonstrated on Table 12.1-13 (Reference 21). 1 77 82

A breakdown of the sources of exposure by work function is presented in Table 12.1-14. This table is based on operating experience as reported in the literature of References 22 through 28, and is presented as fractions of the total man-rem accumulated for 1974. The data have been averaged and evaluated for each work function in order to arrive at the estimated breakdown of the man-rem to be received at ZPS-1. 1 75

Contracts will be administered through the operating life of the plant for services as follows:

a. disposal of radwaste,

b. supply of natural uranium,

c. conversion of natural uranium to UF_6,

d. enrichment of UF_6,

e. fabrication of fuel elements, and

f. handling and reprocessing of spent fuel.

Neither the radwaste disposal (item a) nor the spent fuel handling and reprocessing contract (item f) have as yet (April 1981) been established. The contract for item a through f will be negotiated and administered within Nuclear Operations. Initial contracts for services have been awarded as follows: | 72

a. natural uranium ore: The Anaconda Company (9-year supply); | 82

b. conversion of natural uranium to UF_6: Allied Chemical Corporation (5-year period); | 50

c. enrichment of UF_6: DOE (10-year period);

d. fuel fabrication, first core: General Electric; and

e. fuel fabrication, nine reloads: Exxon Company.

It is contemplated that future contracts will be negotiated, awarded, and administered by the ZPS-1 staff, under the direction of the Plant Manager to cover landscaping, security, miscellaneous supplies, and similar items. | 72

Nuclear Operations will negotiate and administer contracts for capital improvements, revisions, or replacements to the plant, and contracts with architect-engineers, as required, for performance of services during the operating life of ZPS-1. | 72

13.1.1.4 Applicant's Technical Staff

The Applicant's corporate technical staff specifically supporting the operation of ZPS-1 comprises personnel from Generation Construction, Licensing and Environmental Affairs, and Nuclear Operations. Additional support from outside CG&E may be supplied by contract, as required. | 50 | 72

The organizational structure of this technical staff is included in the CG&E Organization Chart (Figure 13.1-1).

The duties, responsibilities, and authorities of the assigned engineering

technical staff, including numbers of personnel, are described in Subsection 13.1.1.2. Qualification sheets for the corporate management positions indicated in Figure 13.1-1 are provided in Attachment 13B. The Applicants arr prepared to furnish resumés of any other members of the current corporate organization, as may be required by the NRC, at any point of time during the period covered by the operating license for ZPS-1. 82

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13.1.2 ZPS-1 Operating Organization

13.1.2.1 Plant Organization

The station organizational chart for Wm. H. Zimmer Nuclear Power Station, Unit 1, is shown in Figure 13.1-2. The principal functional divisions under the direct supervision of the Plant Manager are the operating, maintenance (electrical, mechanical and instrumentation and control), rad-chem and technical divisions and the administrative section. Services such as clerical and purchasing and stores are provided by those groups as indicated in the overall organizational chart. Position titles, numbers of personnel, NRC licenses required and lines of functional reporting, as well as direct lines of communications, are clearly indicated on the organizational chart. 72 81

13.1.2.1.1 Technical Services Division

The Technical Services Division is responsible for all staff support activities including station nuclear engineering. This division is made up of professional personnel with degrees from accredited colleges in various fields of engineering and science. 81

13.1.2.1.2 Operating Division

81

The Operating Division is responsible for all operating functions within the station, with the exception of the liquid and solid radwaste facility (Subsection 13.1.2.1.2). The Operating Division provides operating personnel for the day-to-day routine operation of the station as well as personnel for the preoperational test program, fuel loading, and startup test programs. 81

The operating personnel are divided into shift crews to adequately provide 24-hour, 7-day/week coverage. Each crew consists of a senior licensed Shift Supervisor (SRO), a second senior licensed Operator (SRO), two Reactor Operators (RO) with reactor operator licenses, two Nuclear Plant Operators, and a Shift Technical Advisor to provide advice and counseling to shift personnel.

13.1.2.1.3 Radiation/Chemistry Division

81

The Radiation/Chemistry (Rad/Chem) Division is responsible for all radiation protection activities, chemical control, liquid, and solid waste management. The Rad/Chem personnel: carry out the tasks required to implement the radiation protection program described in Section 12.3; perform the routine and special chemical, radiochemical, and counting analyses required to ensure compliance with the operating license and Technical Specifications and to maintain and improve plant efficiency; and operate the liquid and solid radwaste facilities.

82

TABLE 15.1.31-1

LOSS OF INSTRUMENT AIR SYSTEM - PARAMETERS

TO BE TABULATED* FOR POSTULATED ACCIDENT ANALYSES

	CONSERVATIVE (NRC) ASSUMPTIONS	REALISTIC (CONSERVATIVE ENGINEERING) ASSUMPTIONS
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level corresponding to 105% NBR steam flow	NA	2533 MWt | 73
B. Burnup	NA	NA
C. Percent of fuel perforated	NA	Zero
D. Release of activity by nuclide	NA	NA
E. Iodine fractions (organic, elemental, and particulate)	NA	NA
F. Reactor coolant activity before the accident	NA	NA
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate	NA	NA
B. Secondary containment leak rate	NA	NA
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies	NA	NA
E. Recirculation system parameters (flow rates vs. time, mixing factor, etc.)	NA	NA | 73
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	Subsection 15.1.31.1.1
III. Dispersion Data		
A. Exclusion radius and LPZ distances	NA	NA
B. χ/Q's (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days)	NA	NA
IV. Dose Data		
A. Method of dose calculation	NA	NA
B. Dose conversion assumptions	NA	NA
C. Peak (or f(t)) concentrations in containment	NA	NA
D. Doses	NA	Zero

*As applicable to the event being described.

TABLE 16.3.2-8 (Cont'd)

	TRIP FUNCTION	TRIP SETTING	MINIMUM NUMBER OF OPERABLE INSTRUMENT CHANNELS PER TRIP SYSTEM	REQUIRED ACTION WHEN MINIMUM CONDITIONS FOR OPERATION ARE NOT SATISFIED	REMARKS
8.	Emergency bus power monitor relay	N/A	1	See note 1	Monitors availability of auxiliary power
9.	Pump bus power monitor relay	N/A	1	See note 1	Monitors availability of pump bus power
10.	Pump minimum flow bypass	700 ($\pm$120) GPM	1	See note 1	Minimum flow permissive

NOTES:

1. If the minimum number of operable instrument channels is available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits. If the channel cannot be tripped by the means stated above, that channel shall be made operable within 24 hours or an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. If the minimum number of operable channels is not available, the system is considered inoperable and the requirements of Specification 16.3.5 apply.

in Table 16.4.8-3. Both the particulate filters and charcoal cartridges will be collected weekly from each sampler and analyzed as shown in Table 16.4.8-3.

B. External gamma exposure will be monitored by thermoluminescent dosimeters. Thermoluminescent dosimeters will be collected quarterly to determine the integral exposure rate at each location. Locations of the monitoring stations are shown in Table 16.4.8-3.

2. The annual dose or dose commitment to any organ of an individual in an unrestricted area, from all radioactive iodine and radioactive material in particulate form, shall not exceed 15 mrem.

2. Terrestrial Monitoring Program

Media directly or indirectly related to the human food chain will be sampled and analyzed. The selected media monitoring are green leafy vegetables, meat, milk (dairy animals), wildlife, groundwater, and soil.

Sampling locations, collection frequency, and the corresponding type of analyses and frequencies are given in Table 16.4.8-3.

3. Aquatic Monitoring Program

Samples of surface water, fish, and sediments, benthic organisms and slime will be collected according to the schedule given in Table 16.4.8-3 at corresponding locations. Type of analysis and frequencies are also given in Table 16.4.8-3.

CHAPTER 17.0 - QUALITY ASSURANCE

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17.2 QUALITY ASSURANCE PROGRAM FOR STATION OPERATION

The following is a description of the quality assurance (QA) program which has been established by The Cincinnati Gas & Electric Company (CG&E) and how it provides assurance that the operation of the Wm. H. Zimmer Nuclear Power Station, Unit 1 (ZPS-1), conforms to applicable regulatory requirements and with the design bases specified in the license application.

The ZPS-1 QA program complies with the requirements which are contained in 10 CFR 50, Appendix B, utilizes the guidance provided by the "Orange Book" dated October 26, 1973, and complies with the guidance contained in ANSI N18.7-1976. Additionally, the operating QA program complies with Regulatory Guides 1.8, 1.30, 1.37 (Revision 0), 1.38, 1.39, 1.54 (Revision 0), 1.58, 1.64 - Rev. 1, 1.74, 1.88, and 1.94 which endorse certain ANSI standards referenced in N18.7-1976. |77

The Wm. H. Zimmer Nuclear Operations Manual (the Manual) contains a detailed description of the policies, program element requirements, procedures and implementing responsibilities of the ZPS-1 operations QA program. |82

17.2.1 Organization (Criterion I)

17.2.1.1 General |15

ZPS-1 is owned by The Cincinnati Gas & Electric Company (CG&E), Columbus and Southern Ohio Electric Company (C&SOE), and the Dayton Power and Light Company (DP&L). CG&E, on its own behalf as to its own interest, and as agent for C&SOE and DP&L as to theirs, respectively, is responsible for operation and maintenance of the station including the attendant quality assurance program for station operation. In certain activities, portions of the QA program may be implemented by other organizations, but Cincinnati Gas and Electric Company shall retain the ultimate responsibility for the QA program and shall provide QA verification and audits of safety-related activities of these organizations to ensure that the requirements of the QA program are satisfied. |72

17.2.1.2 Corporate Organization

17.2.1.2.1 Senior Vice-President - Engineering Services and Electric Production |50 |72

The Senior Vice-President - Engineering Services and Electric Production (Senior Vice-President) of CG&E has the ultimate responsibility to the Nuclear Regulatory Commission (NRC) for the engineering, design, procurement, construction, and operation of ZPS-1 and for ensuring quality in these phases of the station's life. |50 |72

The responsibility for quality assurance is implemented through the establishment of policies and practices in the QA manual which are

applicable to the quality assurance program. The QA manual shall be finally approved by the Senior Vice-President, who shall ensure that the independence of the Manager - Quality Assurance, in matters relative to quality assurance, exists between operational, scheduling, and cost considerations. The Manager - Quality Assurance shall have sufficient organizational freedom and authority to direct and control quality assurance in accordance with the requirements of the QA program. 50 72

In addition, the Senior Vice-President shall review independent quality audits of operations phase activities and shall ensure that corrective action as required is implemented. The responsibility for all audits, other than of QA, is delegated to the Manager - Quality Assurance; however, the Senior Vice-President appoints members to the CG&E operations review committee (ORC) and shall be its chairman and shall be responsible for the activities of this group. As such, the Senior Vice-President retains the authority to conduct or order an audit of any activity at any time to determine compliance with or the effectiveness of the policies and practices he has established in the QA program. 50 72 72 50

The Senior Vice-President shall be responsible for ensuring that an annual review of the QA program is conducted and for maintaining formal communication with the NRC on quality-related matters. 50

17.2.1.2.2 Operations Review Committee (ORC) 15

The ORC is a group of individuals appointed by the Senior Vice-President, in his capacity as ORC Chairman. 50 72

The primary responsibilities and membership of the ORC are described in Subsection 13.1.1.1.2. 72 82

17.2.1.3 Quality Assurance

Figure 17.2-1 shows diagrammatically that Quality Assurance (QA) is independent of the individual or group directly responsible for performing the specific engineering activities during the operational phase of the facility.

The Manager - QA is responsible for administration of quality assurance. The Manager - QA reports to the Vice-President - Nuclear Operations for all quality assurance matters but maintains an open line of communication to the Senior Vice-President. These quality assurance matters include, but are not limited to:

a. supervision and administration of his QA staff including selection and salary review of personnel assigned to positions within QA;

b. preparation, review, and approval of QA procedures to implement the policies and responsibilities detailed in the Nuclear Operations Manual. These procedures shall include requirements to ensure that updated codes, standards, and applicable regulatory guides have been appropriately addressed;

c. review and approval of audit schedules, supervision of the planning for audits, review of audit results, and assurance that followup of nonconformances is accomplished;

d. reporting audit results, nonconformance, noncompliance with procedures, and other quality conditions which he considers necessary to the vice president to ensure the effectiveness and adequacy of the QA program;

e. coordinating quality activities with the Plant Manager; and

f. initiating stop work action through appropriate channels or controlling further processing, delivery, or installation of nonconforming components or material during periods of major station modification or repair.

During major modifications and/or major repairs that affect design of essential structures, systems, or components, QA has the authority to initiate stop work on those repairs and/or modifications. This authority will be procedurally controlled.

17.2.1.4 The Wm. H. Zimmer Nuclear Power Station

The Plant Manager is responsible for the safe, reliable, and efficient operation of ZPS-1, and for adhering to and for the

implementation of pertinent CG&E management directives to be applied by the Plant Manager to the nuclear quality-related activities of the station. 72

Other direct responsibilities of the Plant Manager include: 72

a. delegation of specific responsibilities to key station management, supervisory, or technical personnel;

b. chairmanship of the station review board;

c. approval of station administrative control documents;

d. approval of the licensed operators requalification program;

e. ensuring enforcement of the station radiation protection program and policies; 72

f. approval of the station internal audit program;

g. approval of all site QA/QC procedures; 15

h. stop work authority on any site-related activity, which includes removing ZPS-1 from service; and 72 82

i. evaluating the operating QA program for adequacy and status, documenting the history of the program, and reporting the findings to company management.

Key personnel comprising the ZPS-1 staff are described in detail in Chapter 13.0. The responsibilities and qualifications of these personnel are also outlined in that chapter; however, certain key group leaders and some of their responsibilities follow. 72 82 72

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Superintendent - Technical Services - The Superintendent - Technical Services is responsible for supervision of the station reactor engineer and the station staff engineering group in the specialized technical areas of nuclear engineering, computer engineering, fuel management and accountability, special studies and test. Other specific responsibilities include: 82

a. ensuring that technically correct procedures are developed for activities within his jurisdiction, such as nuclear engineering and the plant computer;

b. ensuring that the need for special fuel handling procedures is identified and that the procedures are prepared; 15

c. ensuring that operating or other station procedures that may be assigned to his group are prepared and reviewed;

d. ensuring that accountability is maintained for all core components;

e. preparing and/or reviewing technical reports or documents resulting from studies or tests performed by his assigned personnel;

f. preparing abnormal occurrence reports including the necessary evaluations, recommendations, or solutions to prevent recurrence; and

g. supervising the establishment and maintenance of the necessary files for documentation of his section's activities. 82

Superintendent - Operations - The Superintendent - Operations is responsible for supervising and coordinating all station operations in accordance with the operating license and technical specifications addressed thereto. Specific responsibilities include:

a. ensuring that procedures are prepared and provided for operation and surveillance of safety-related mechanical and electrical components and systems, including integrated system operation;

b. periodically reviewing control room logs and operating procedures to ensure adherence to established routines and instructions;

c. preparing memorandums pertaining to plant operation and reviewing significant operating events or other unusual operating events;

d. participating in the operator training, retraining, and requalification program; and 15

e. supervising the establishment and maintenance of the necessary files for documentation of his section's activities. 82

Superintendent - Maintenance - The Superintendent - Maintenance is responsible for all mechanical and electrical maintenance activities including routine inspections, preventive maintenance, repairs, and required record keeping, all in accordance with established administrative policies. Additional responsibilities include:

a. ensuring that procedures are prepared, reviewed, approved, and followed for activities related to his responsible area;

b. ensuring station instrument and control maintenance, calibration, test, and surveillance;

c. ensuring that the above activities designated safety-related are performed in accordance with the latest approved revision of the applicable procedure; 82

d. ensuring that measuring and test equipment used during safety-related maintenance and I&C activities are properly controlled and calibrated.

e. coordinating maintenance and surveillance activities with the operating, rad/chem, maintenance, and station quality engineers as necessary and required;

f. coordinating major maintenance or modification work activities with contractors, QA, and GED as required;

g. procuring essential spare parts in accordance with approved procurement procedures; and 82

h. supervising the establishment and maintenance of the necessary files for documentation of his section's activities.

Superintendent - Radiation/Chemistry - The Superintendent - Radiation/Chemistry is responsible for all station radiation protection, chemistry, radiochemistry, and radwaste activities to support the activities of the entire station. Additional responsibilities include:

a. ensuring that procedures are prepared, reviewed, approved, and followed for activities related to his area of responsibility; 15

b. preparing implementing procedures in support of the site emergency plant;

c. training and retraining plant personnel in sound radiation protection practices and in emergency practices and procedures;

d. controlling all radwaste discharge from the station; and

e. supervising the establishment and maintenance of the necessary files for documentation of his section's activities. 82

Office and Document Control supervisor - The office and document control supervisor is responsible for supervising and coordinating the activities of the clerical staff to support the activities of the entire station. Additional responsibilities include:

a. ensuring that procedures are prepared, reviewed, approved and followed for implementing activities in his area of responsibility;

b. indexing, controlling, and issuing plant procedures, tests, directives, memorandums, and instructions; and

c. indexing and controlling station records as specified in the approved administrative control directive.

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17.2.1.5 Station Review Board

The membership, responsibilities, and authorities of the station review board (SRB) are set forth in the Station Technical Specifications, Section 6.5.1. The SRB is responsible for review of certain safety-related procedures, tests and test data, abnormal occurrences, security plan, emergency plan, etc.

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The SRB is concerned with the technical aspects of the station but shall coordinate its activities with QA to ensure that applicable quality aspects of station operation and maintenance are satisfied.

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17.2.2 Quality Assurance Program (Criterion II)

The quality assurance (QA) program shall be in effect at all times during the operational phase of ZPS-1 to ensure that activities performed during this portion of station life are carried out without undue risk to the health and safety of the public. The QA program requires the establishment of management controls as hereinafter specified such that decisions affecting safety are made at the proper level of responsibility and with the necessary technical advice and review.

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The QA program recognizes that QA involves many organizations and individuals and is therefore not regarded as the sole responsibility of a single QA group. The QA program extends to various levels in all participating organizations, from the company president to all workers whose activities may affect or influence quality. |82

The QA program shall apply to all activities affecting the safety-related functions of structures, systems, and components. These activities include design changes, purchasing, fabricating, handling, shipping storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. |82

The safety-related activities associated with station operation fall into several major categories, as follows:

a. Station tasks that exclusively involve or are the responsibility of the ZPS-1 station staff. Activities such as, but not limited to, operation, maintenance, surveillance, testing, calibration, inspection, refueling, and record keeping are in this classification.

b. Items such as major modification and major repair that affect design of structures, systems, or components are included in this grouping. This effort includes the design, specification preparation, review, equipment purchase, contractor/consultant liaison, and attendant QA program that governs these tasks. Also included in this activity listing is the design coordination, yellow-cake purchase, conversion, enrichment, fabrication, and delivery of all reload fuel batches. |72 15

c. Interfacing or coordinating efforts that involve the station staff and other organizations as necessary or required by the QA program. Examples of these activities include independent review and audit of station operation; coordination and interface during major modification, and other design or repair tasks; construction testing and startup testing; refueling; purchase of essential or safety-related spare parts and equipment; and licensing and environmental affairs. |72

Other items having important reliability considerations for achieving long-life, trouble-free service but not having a prime safety function should be included in the QA program. The QA coverage for these components and systems is based on the guidance provided in the management directives issued by the plant manager as they apply to his scope of responsibility; and upon the policy, experience, and direction of the Vice-President, as they apply to major design changes and/or modification. |82 |72 |82

Provisions are established for the resolution of disputes involving quality and arising from a difference of opinion between QA/QC personnel and other department personnel such as engineering, procurement, vendors, etc. The organizational table presented in Figure 17.2-1 shows the lines of responsibility and communication established for effective implementation of the QA program. Procedures will specify that the resolution of conflicts follow the lines of responsibility designated in Figure 17.2-1. When necessary, these conflicts are taken to the highest level of management necessary for resolution. 82 82

17.2.2.1 Policy

The management of the Cincinnati Gas & Electric Company recognizes their responsibility to establish controls and policies which will ensure that the design, construction, and operation of ZPS-1 is conducted in such a manner as to prevent conditions which may be adverse to the health and safety of the public.

The Nuclear Operations Manual (the Manual) is a management policy document which is authorized, controlled and is the responsibility of the president. The Manual formally establishes the management policy for quality assurance, describes the QA program, and establishes requirements and guidelines for the written procedures required to implement the QA program. The Manual also serves to inform those company groups, departments, or sections and other outside contractors, consultants, and participants about their interactions and responsibilities as they pertain to quality. 82 15

The primary elements of the QA program established in the Manual are administrative controls, reviews, and audits, reporting requirements, management review, and indoctrination and training.

17.2.2.2 Structures, Systems, Components, Activities and Procedures

Significant structures, systems, components, activities, and procedures have been reviewed to evaluate their safety-related functions. Safety-related equipment and procedures fall into the following categories.

17.2.2.2.1 Safety-Related (Essential) Components

Essential components or items are characterized by important nuclear safety-related functions.

Safety-related is defined to mean those structures, systems, or components necessary to meet any or all of the following:

a. Maintain the integrity of the reactor coolant pressure boundary. 82

b. Provide the capability to shutdown the reactor and maintain it in a safe shutdown condition, per 10 CFR 100.

c. Provide the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100. 82

A Q-list has been prepared under the authority of the Manager-NED, that lists all essential components utilized in ZPS-1.

17.2.2.2.2 Safety-Related Activities

A safety-related activity is defined as any activity which may affect the ability of a safety-related structure, system, or component to fulfill its intended function. Further, an activity itself, such as manual initiation of engineered safety equipment, which could mitigate the effects of potential accidents, is considered a safety-related activity (Table 17.2-1). 82

17.2.2.2.3 Safety-Related Procedures

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Safety-related procedures are defined as written procedures or instructions defining the normal method, means, and limits of executing safety-related activities (see Chapter 13.0). 72

17.2.2.3 Organizations and Functions

The QA program involves several organizations and individuals, including the following.

17.2.2.3.1 Quality Assurance 72

The Senior Vice-President, in accordance with company policy, has designated the Manager-Quality Assurance, as the responsible management representative for ZPS-1 quality assurance.

The Quality Assurance Department has overall responsibility for the quality-related aspects of design, construction, and operation of the station. 82

These responsibilities include the establishment, in accordance with the Manual, of a QA/QC program which governs the safety-related activities of the station technical staff, technicians, subcontractors (if necessary), and vendors of materials, spare parts, or equipment procured by the plant.

An additional responsibility assigned to QA during station operation includes independent review or audit of station operation. This ensures that a deliberately critical examination, evaluation of audit results, procedures, certain contemplated actions, and after-the-fact investigations are conducted by personnel not having direct responsibility for the work function under review. The audit program shall be carried out to verify compliance with all aspects of the administrative controls and QA program and to ensure that an audit of all safety-related activities is completed within a period of 2 years. 72 72 15

17.2.2.3.2 Nuclear Production Department

The Manual assigns the Plant Manager overall responsibility for the conduct of station operations. Included in this responsibility is the requirement to ensure that safety-related activities are conducted in accordance with the requirements of the QA program.

Additionally, normal review, checking, and verification of station activities shall be performed by both first-line and second-line supervisory and/or technical station personnel. Attainment by the QA programs objectives is also accomplished to those who have been assigned responsibility for performing work. This may include interim examinations, checks, and inspections of the work by the individual performing the work. 82

17.2.2.3.3 Quality Assurance Interfaces

CG&E is responsible and answers to the NRC Office of Inspection and Enforcement and to the State of Ohio. 72

QA questions on design and specifications are handled through project sponsors in CG&E.

If QA problems of major consequence become apparent, QA has the authority to contact appropriate people in other involved organizations directly. 72

QA may engage special consultants to conduct inspections, tests, or audits or to duplicate or otherwise verify the regular inspections performed in the station by the station personnel. 72

QA coordinates and communicates information on QA matters. 72

Procedures require that any flow of QA information between CG&E organizations or any of their contractors, subcontractors, consultants, agents, or other involved parties pertaining to plant quality shall be routed through QA. QA is responsible for QA contacts with vendors furnishing essential components or services under CG&E purchase orders. Any direct contact necessary to expedite the flow of information shall have had prior clearance by QA and shall be documented with a copy to QA. 15 72 72

17.2.2.4 Administrative Controls

The QA program consists of managerial, administrative, and operational controls and procedures used to ensure the safe operation of the facility. These documented procedures and controls are contained in several diverse documents, but when taken together, these documents completely describe the QA program in that their collective implementation assures and provides adequate confidence that structures, systems, or components will perform satisfactorily in service.

The Manual requires, as part of the total QA program, the preparation of appropriate administrative controls which prescribe the criteria established to control all safety-related activities. The general listing of safety-related activities for which administrative controls shall be prepared have been previously listed in Table 17.2-1. The definition of these activities is aimed at administrative responsibility and/or the functional nature of the nature of the activity. 15

Redefinition of this list may be expected to occur as a result of management reviews of the adequacy of the QA program over the station's life.

The administrative control documents establish measures to control and coordinate the approval and issuance of documents, including changes, which prescribe all safety-related activities. Such documents include those which describe organizational interfaces; assign authorities and responsibilities; and prescribe activities affecting essential structures, systems, or components.

Each activity is controlled to comply with the appropriate requirements of 10 CFR 50, Appendix B. Compliance to Appendix B requirements is the responsibility of the approval authorities having jurisdiction for the particular activity, while implementation of the requirements is the task of the individual or group with primary responsibility for that activity.

These documents also include the requirement that safety-related activities be accomplished according to written procedures of a type appropriate to the circumstances and shall be adhered to. A listing of safety-related procedures has been previously documented in Chapter 13.0. The QA program complies with the guidance given in ANSI N18.7-1976. Procedures include operating and maintenance procedures, material control procedures, surveillance procedures, inspection and test procedures, instrument calibration procedures, chemical/radiochemical procedures, emergency procedures, and other implementing procedures as may be identified or required. Provisions and measures shall be established assuring that procedures, including revisions thereto, are reviewed for adequacy by technically competent individuals or groups, are approved for release by authorized personnel, and are distributed in accordance with current distribution lists. These controls provide assurance that the procedures are used by the personnel performing the prescribed activity and that procedures are established to avoid the use of incorrect or outdated documents. A matrix will be developed listing each criterion of Appendix B to 10 CFR 50 that references detailed procedures which implement the criteria. | 82 | 15 | 82 | 82

17.2.2.5 Reviews and Audits

The review and audit program shall be established to verify that safety-related activities are performed in accordance with the requirements of the QA Program. These review and audit programs shall themselves be periodically reviewed for effectiveness by the vice president or his designated representative. | 82

Review and audit programs shall be preplanned, written documents that contain: subjects to be audited and independently reviewed; responsibility and authority of those supervising audits and conducting reviews, including responsibility to identify problems and recommend solutions to appropriate levels of management; mechanisms for initiating audit and review activities; provisions for use of specialists or subgroups; | 82

authority to obtain access to station files and personnel; requirements for distribution of reports; identification of the management position to which auditors and independent reviews report; and provisions for followup action, including reaudit of deficient areas.

17.2.2.5.1 Independent Safety Review Group (ISRG)

Independent review programs shall be accomplished utilizing the ISRG. ISRG functions and responsibilities are detailed in FSAR Chapter 13.0 and the Station Technical Specifications. 72

17.2.2.5.2 Operations Review Committee (ORC) 82

The ORC is a group of individuals, appointed by the Senior Vice-President in his capacity as chairman of the ORC, whose primary responsibility is to identify and evaluate major risks and make recommendations on how to resolve them. ORC functions and responsibilities are detailed in FSAR Chapter 13.0 and the Station Technical Specifications.

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17.2.2.5.3 Station Review Board (SRB)

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The SRB functions to advise the Plant Manager on all matters related to nuclear safety. The SRB composition, authority, responsibilities, function, and meeting frequency are detailed in the Station Technical Specifications. Some of the review functions of the SRB are:

a. review of all proposed safety-related plant procedures,

b. review of all tests and experiments that could affect nuclear safety,

c. review of proposed changes to technical specifications, 15

d. review of proposed changes or modifications to plant systems or equipment that affect or could affect nuclear safety,

e. review of plant operation to detect potential safety hazards,

f. performance of special reviews and investigations as requested by the chairman of the ORC, and

g. review of the station security and emergency plans and implementing procedures addressed thereto.

Administrative controls shall be established to govern the activities of the SRB. | 82

17.2.2.5.4 Quality Assurance (QA) | 72

The Manager - QA shall be responsible for directing QA personnel in the performance of planned and documented audits to verify compliance with all aspects of the operation QA program.

Audits should include verification of compliance and effectiveness of implementation of internal rules, procedures, regulations, and license provisions; programs for training, retraining and license operator requalification; corrective actions taken following abnormal occurrences; and record management systems. Written reports of such audits shall be reviewed by the ORC and by appropriate members of management including those having responsibility in the area audited.

17.2.2.5.5 Internal Station Audit | 82

Special reviews or internal audits should be periodically performed by members of the station staff. These audits should be scheduled on a not-to-interfere basis and should be scheduled with the supervisor of the group or activity audited. These internal audits shall be the responsibility of the Plant Manager and shall be scheduled in accordance with the minimum frequency stipulated by him. | 72 | 82

These audits should provide a continuing evlauation of plant operation under the jurisdiction of the Plant Manager and an aid in planning future activities and/or corrective action as may be necessary. The important elements of the reviews should be documented. | 15 | 72

The Plant Manager shall be responsible for timely referral of appropriate matters to management and independent reviewers. | 72

17.2.2.5.6 Station Operating Organization | 82

The station managerial, supervisory, and technical staff shall provide, as part of their normal duties, timely and continuing monitoring of operating activities to assess the plant conditions and to verify that day-to-day activities are conducted safely and in accordance with applicable administrative controls.

17.2.2.6 Indoctrination and Training

A training program shall be established that provides training for personnel performing activities affecting quality commensurate with the importance or functional nature of the activity and to ensure that suitable proficiency is maintained. The program shall also cover areas such as: company policies, administrative controls, and procedures or instructions which implement the program related to the specific activity. | 72

17.2.2.7 Preoperational Testing

The operational phase of station operation is generally considered to commence with initial fuel loading, except for certain preoperational activities. No clear dividing line can be established that adequately describes when construction activities cease and operating activities begin, because certain operating functions start prior to fuel loading, while certain construction activities may extend beyond fuel loading.

An administrative preoperational and startup manual shall be prepared to clearly and specifically identify the activities that fall within these overlapping time periods and to specify authorities and responsibilities for the various construction tests, inspections, verification, preoperational tests, and startup tests. Appropriate procedures shall specify tasks such as construction tests, turnover and acceptance, preoperational tests, startup, and power tests. Responsibilities and authorities, QA requirements, approval authorities, system boundaries, maintenance requirements, necessary surveillance, water quality requirements, identification and resolution of deficiencies, safety tagging, etc., shall be detailed in the startup manual.

17.2.3 Design Control (Criterion III)

The quality assurance manual for station operation requires that any design activities during the operations phase, to the extent that they affect nuclear quality-related functions of structures, systems or components, are controlled by procedures that are established to carry out those activities in a planned, controlled and orderly manner. These procedures contain design controls that are applied to such disciplines as reactor physics; seismic, stress, thermal, hydraulic, radiation, and accident analyses; compatibility of materials; and accessibility for inservice inspection, maintenance, and repair. The use of the procedures by CG&E personnel and/or their contractors or consultants ensures that applicable regulatory requirements, design bases, and appropriate quality standards are correctly translated into design documents such as specifications, drawings, written procedures and instructions, and deviations and changes from such standards are controlled. Design control includes provisions which ensure that: 15 82

a. The selection of suitable materials, parts, equipment, and processes for nuclear quality-related structures, systems and components includes the use of current industry standards and specifications and material and prototype hardware testing programs.

b. Standard "off the shelf" commercial or previously approved materials, parts, and equipment that are essential to the nuclear quality-related functions of structures, systems or components are reviewed for suitability of application prior to selection. 82

c. When more than one organization is involved, interface controls are described for all participating organizations.

The Manual requires that individuals or groups responsible for design verification have the qualifications to review and are other than the original designer. These individuals or groups responsible for independent design verification and their authority and responsibility are established and controlled in administrative procedures.

Procedures controlling design also establish criteria for the proper selection and accomplishment of design verification or checking methods such as design reviews, alternate calculations, and/or qualification testing. Design verifications are accomplished to provide assurance: |82

a. That studies, calculations, and analyses involving nuclear effects, electrical, mechanical, thermal, reactor physics, hydraulic and other pertinent considerations are complete and correct.

b. That maximum use is made of qualified and approved parts, material or components.

c. That the proper design verification method was used to check the design and that it was performed by a sufficiently independent individual or group.

d. That tests associated with design changes prove the materials and design interfaces are compatible.

e. That acceptance criteria for inspection and test are delineated and effective. |15

f. That accessibility for in-service inspection, maintenance or repair is adequate.

When a test program is used to verify the adequacy of design, a qualification test of a prototype unit under adverse design conditions is used.

Any errors or deficiencies found during design verification or actual operation that could adversely affect nuclear quality-related structures, systems, and components are documented in accordance with administrative and quality assurance procedures which also define the types of corrective action to be taken.

Design documents and revisions thereto are distributed in accordance with the procedures as established in Subsection 17.2.6, and those documents, reviews, records and changes are collected, stored and maintained in accordance with the procedures established in Subsection 17.2.17.

All records for design and specification modification initiated by station personnel that affect nuclear quality-related function of systems, structures, and components are identified by station administrative procedures and forwarded to the Nuclear Engineering Department for detailed design work, review and approval. All design and speci- |82

fication changes are subject to the same design controls commensurate with the original design and are approved by the organization that performed the original design or other organization qualified to perform such review.

The Vice-President - Nuclear Operations has the overall responsibility for ensuring that design control procedures are prepared and implemented incorporating the design control practices delineated previously, whether the work is performed by the Cincinnati Gas & Electric Company or other organizations. He is also responsible for the timely approval and updating of specifications, drawings, installation requirements, and other design documents, utilized for purchase or installation of materials, parts or components, and for the maintenance of those documents in accordance with the requirements for document control and records management. In addition, he ensures that the necessary criteria for the proper selection of the method of design verification are used. These control criteria are incorporated into the administrative procedures. |72|82

The Manager - QA is responsible for approving the administrative procedures which control design. He reviews those procedures to ensure they meet all quality assurance requirements established in the Manual. |72

For all nuclear quality-related equipment, Quality Assurance Department (QAD) personnel audit reviews of design analyses for compliance with design criteria. These audits ensure that design characterization has been controlled, inspected, and tested and that inspection and test criteria are identified. |82 15

Specifications, design documents, drawings, installation requirements, and changes thereto for nuclear quality-related structures, systems and components are reviewed by QAD personnel to ensure that quality assurance/quality control requirements, criteria, records, and other required documentation have been included in the design document. |82

17.2.4 Procurement Document Control (Criterion IV)

The administrative control procedure established for the procurement of essential spare parts, components, or materials will specify that quality requirements at least equivalent to those stipulated in the original purchase specification be included on the purchase order. The administrative procedure will require that documentation such as drawings, specifications, procedures, inspection plans, test records, qualification records, material records, etc., be identified as applicable for the purchased items and included on the purchase order. Additionally, instructions to the contractor will require the submittal of this documentation to the purchaser for review and approval.

In addition, the procurement documents are required to identify the applicable requirements of 10 CFR 50, Appendix B, with which the supplier QA program must comply as well as the document requirements

for drawings, specifications, procedures, personnel qualifications, procedure qualifications, material test results, and inspection and test records which must be prepared, maintained, submitted, or made available for review and/or approval. The requirements for retention, control, maintenance, or delivery of the applicable records as well as the purchaser's right of access to the vendor's facilities and/or records for source inspection and/or audit are specified. The procurement documents are subject to QA program requirements equivalent to those specified for original equipment purchases, or those specified by properly reviewed and approved revisions.

The Materials Management Department of CG&E is responsible for the processing of purchase orders. The organization originating the purchase request (NPD, NED, NSD, or QAD) is responsible for incorporating or assuring that the documents contain the applicable technical and quality requirements. Authorized release of the procurement documents is required prior to placing a purchase. For purchase documents generated by ZPS-1 station personnel, the technical staff personnel are responsible for issuing requests to the Stores & Salvage Division. QA determines if measures have been established to ensure that applicable regulatory requirements and design bases and other requirements are suitably included or referenced in the documents for procurement of essential material, equipment, or services. QA is responsible for review of the purchase documents for inclusion of appropriate quality requirements, and the Plant Manager, or designee, is responsible for approval prior to release from the station. Additionally, QA is available for review of quality requirements as may be needed.

Purchases made for materials and components (major design changes or modifications) are reviewed and approved as specified in Subsection 17.1.4.

Agents for CG&E such as consultants, constructors, testing companies, laboratories, architect-engineers, etc., delegated the responsibility for procurement services associated with essential (safety-related) structures, systems, components, or services shall be required to meet all of the applicable criteria outlined above.

17.2.5 Instructions, Procedures, Drawings (Criterion V)

Administrative control procedures are prepared, reviewed, and approved under the direction of the Vice-President - Nuclear Operations to implement the quality assurance program. QA reviews and concurs with all administrative control procedures affecting safety-related activities to assure that necessary quality requirements are prescribed.

Activities affecting the quality of the operation of the station shall be performed in accordance with written instructions, procedures, and drawings under the cognizance of the Plant Manager. These documents prescribe acceptable methods for implementing these activities and refer to appropriate inspections and

tests. Acceptance criteria shall be established which are the basis for assuring that the performance of the activities is in accordance with the instructions, procedures, and drawings which are applicable to the activity. Typical activities which include requirements for quality are operation, testing, maintenance, and personnel qualification procedures. In specific instances concerning the sequence of actions for the preparation, review, and approval chain for station instructions and procedures, the administrative procedure will establish those necessary requirements to assure that the review and approval of procedures or instructions are controlled. The administrative procedure will clearly identify the supervisor delegated the responsibility for ensuring that procedures are prepared for safety-related activities concerning areas within their purview (see also Chapter 13.0 and Subsection 17.2.1.5 for a description of key supervisory personnel responsibilities). | 82

Additionally, the administrative procedure shall include the required approval sequence for each general classification of procedures.

Generally, the supervisor shall be responsible for approval of safety-related procedures within his delegated area, and the station review board shall review and recommend approval/disapproval of all safety-related procedures. The Plant Manager's signature shall indicate final approval and authority to implement the procedure as soon as possible. | 82 | 46 | 72

The administrative procedure for procedure review and approval provides instructions to the review and approval authorities to assure that: 15

a. The procedure or instruction is complete.

b. The intent of the procedure has been adequately covered.

c. Procedures are compatible with similar documents.

d. Technical specifications are addressed as may be necessary.

e. An unreviewed safety question has not resulted.

f. The procedure does not violate documents such as the FSAR, safety analyses, or other controlling documents.

QA reviews and concurs with safety-related maintenance, modification, calibration, receiving, inspection, and special process procedures including revisions or changes thereto to assure that necessary quality requirements are prescribed. | 72

For station activities including for the most part, or falling under the responsibility of other Nuclear Operations' Departments and CG&E Departments, their contractors, and/or consultants (Subsection 17.2.2), the review and approval chain shall be similar to the procedures established during design and construction. | 72 | 82

Instructions, procedures, and drawings are audited by QA personnel to ensure reference to appropriate specifications and codes and stan-

dards requirements. Regulatory guides and state and local requirements are referenced in the documents. The QA personnel audit the results of inspections and tests to ensure conformance with the acceptance criteria established within these documents.

Instructions, procedures, and drawings which are used for plant equipment replacements are controlled in accordance with criteria established for the construction phase of the plant installation.

17.2.6 Document Control (Criterion VI)

The QA program provides for the control of documents, including revisions or changes, which affect the quality of safety-related structures, systems, or components. These documents include, but are not limited to, the following: 82

a. FSAR,
b. Nuclear Operations Manual 82
c. design and installation drawings,
d. manufacturers drawings and instruction books,
e. design specifications,
f. procurement documents,
g. administrative control procedures, and
h. station procedures manual.

The administrative control procedures established for document control identify and control the individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto. 15

QA determines that measures have been established to control the distribution of documents which prescribe activities affecting quality. 72

The above measures ensure that changes to these documents are reviewed for adequacy and completeness and are in accordance with design criteria which are requirements of codes, standards, and applicable regulatory requirements. The control ensures that the documents are distributed in the same manner as the original document. The measures ensure that documents are released for issue only after authorized personnel have reviewed and approved such documents. Revisions to such documents are normally reviewed and approved by the same organization that was authorized to release the original document. Quality assurance personnel have the option to designate another responsible organization other than that which approved the original document to review and approve revisions to the original document if such organization has demonstrated its capability to perform such review.

The document control register index, established during the construction of the plant, is maintained during the operation of the plant. This register records the latest revision of all design documents applicable to the plant installation, operation, and maintenance. The documents listed on the register are available at the plant site are distributed and used by personnel performing activities prescribed

by the appropriate document. QA audits the above measures to ensure that such measures are implemented and functioning and are controlled as intended. 72

17.2.7 Control of Purchased Material, Equipment, and Services (Criterion VII)

The QA program requires that measures which assure that safety-related material, equipment, and services procured during the operating life of the station conform to procurement document requirements. The administrative control procedure establishes the procedural control assuring that spare or replacement parts of safety-related structures, systems, or components are subject to controls equipment. The program provides for source evaluation, selection, inspection, surveillance, and audit, and for receiving inspection of nuclear quality-related materials. 82

17.2.7.1 Source Evaluation and Selection

Potential suppliers of nuclear quality-related items or services are required to submit their QA/QC programs as evidence of their ability to meet purchase order and/or specification requirements. QA may use support from the engineering groups in the engineering department, its QA consultant, or the experience of other agencies knowledgeable as to performance of the supplier. 82 72

The evaluation of suppliers is conducted in accordance with written procedures and is based on one or more of the following: 15

a. the supplier's capability to comply with the elements of 10 CFR 50, Appendix B, ANSI N45.2, and the ASME Code that are applicable to the type of material, equipment or service being produced,

b. a review of the previous records and performance of suppliers who have provided similar articles of the type being produced; and

c. a survey of the supplier's facilities and quality assurance program to determine his capability to supply a product which meets the design, manufacturing, and quality requirements. Results of the surveys are documented and filed at the site document center.

An acceptable QA/QC program is a prerequisite to award of work.

17.2.7.2 Source Inspection, Surveillance, and Audit

The effectiveness of the control of quality by contractors, suppliers, and subcontractors is established by inspection, surveillance, and audit of those suppliers conducted by QA personnel or by its agent. Inspection, surveillance, and audit of suppliers during fabrication, inspection, testing, and shipment of materials, equipment, and com- 72

ponents is planned and performed in accordance with written procedures to ensure conformance to the purchase order requirements. These procedures contain provisions for:

a. instructions that specify the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extend of documentation required; and those responsible for implementing these instructions; and

b. audits and surveillance which ensure that the supplier complies with all quality requirements. Surveillance is performed on selected items where verification of procurement requirements cannot be determined upon receipt.

17.2.7.3 Receiving Inspection

Receiving inspection of nuclear quality-related material is performed in accordance with written procedures to ensure the following:

a. The material, component, or equipment is properly identified and conforms to specific requirements of the procurement document such as codes, standards, or specifications.

b. Packaging and preservation is adequate to ensure that the item will remain in acceptable condition until it is installed or used. 15

c. The material, component, or equipment is inspected and judged, along with the acceptance records, to be accptable in accordance with predetermined inspection instructions, prior to installation or use.

d. Inspection records or certificates of conformance attesting to the acceptance of material, component, and equipment shall be available at the nuclear power plant prior to installation or use.

e. There is a controlled storage area for accepted materials.

f. Those purchased items under the control of either "offsite" or "onsite" organizations are identified and characterized accordingly.

g. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

h. Nonconforming items are segregated or controlled, and clearly identified until proper disposition is made. 82

The Manager - QA has the overall responsibility to audit the control measures used to ensure that the proper receiving inspections are accomplished. QA is responsible for ensuring that the receiving inspections are performed in accordance with the approved, written procedures, and check lists developed for that specific purpose: 72

a. certifications that specifically identify (e.g., by the purchase order number) the purchased material or equipment and the specific procurement requirements (codes, standards, specifications, etc.) met by the items;

b. certification identifying any procurement requirements which have not been met together with a desription of nonconformances dispositioned and approved by those authorized and qualified to "accept as is" or "repair";

c. documents demonstraing the traceability of the items where required by applicable codes; and

d. where qualification testing has been performed on either the purchased item or a prototype, data on the results of the tests that demonstrate the item's acceptability.

QA reviews this documentation in accordance with approved written procedures to ensure compliance with procurement requirements. Audits shall be conducted by QA, a designated QA consultant, or other qualified representative, at intervals consistent with the importance, complexity, and quantity of material or service being supplied to ensure that effective control of quality is maintained by the supplying contractor and subcontractor. 82 72 15

17.2.8 Identification and Control of Materials, Parts and Components (Criterion VIII)

Administrative control procedures establish the requirements for receiving inspections; materials, parts, or component identification and inventory control; storage and preservation; and storeroom requisitions, etc. Included in these administrative control documents are instructions such as:

a. responsibilities and authorities of the storekeeper, QA, group supervisors, or other key personnel involved in the total function of purchasing and stores; 72

b. requirements for marking or identifying materials, parts, or components;

c. inventory control;

d. storage requirements as they relate to the necessity for special protective measures from humidity, temperature fluctuations, dirt, physical damage, etc.; and

e. storeroom requisition procedures.

Contained in these procedures are provisions that consider the method of identification that it does not affect the fit, function, or quality of the item being identified.

The QA program requires that safety-related materials, parts, and components be controlled and identified during the operations phase of the plant. Procedures which involve identification and control of materials, parts, and components contain written provisions which ensure that the item is maintained by heat number, part number, serial number, or other controlled means during storage, fabrication, installation, operation, and maintenance involving these items. Those organizations involved in all design and design change activities shall be responsible for the establishment of identification requirements for the safety-related materials, parts, and components used in the design. Appropriate station personnel shall submit the operational requirements to these design organizations. 82 72

Persons responsible for maintenance activities shall be knowledgeable as to the identification assigned to the individual materials, parts, and components during their maintenance. Appropriate documentation shall be prepared to identify such items during these activities. Engineering and operating personnel shall properly identify these items in procurement documents. Appropriate documentation shall be submitted to ensure the continuous maintenance of the identification. 15

Quality assurance personnel shall review those procedures which contain provisions for identification for materials, parts, and components to ensure that the items can be traced to identifying documents such as drawings, specifications, purchase orders, manufacturing and inspection documents, nonconformance reports, physical and chemical mill test reports, and other similar documents. 82 82

Written procedures define the controls placed on safety-related materials, parts, and components upon receipt, inspection, storage, and issue for use or installation. Items returned to offsite locations for repair shall be controlled to ensure propr identification. Materials management department personnel assigned at the operating plant are responsible for the control of these items. Verification is required that the correct identification of safety-related materials, parts, and components is performed and documented during storage and prior to release for fabrication, assembly, shipping, installation and during operation. This verification, combined with the quarantine of items found deficient during receiving inspection, ensures that incorrect or defective items will not be inadvertently used.

17.2.9 Control of Special Processes (Criterion IX)

The QA program requires that special processes that affect the quality of essential structures, systems and components be qualified and controlled in accordance with written procedures. 82

Persons performing these operations shall be appropriately qualified in accordance with these procedures.

Written procedures are prepared and authorized under the supervision of the Plant Manager. These procedures ensure that special processes such as welding, heat treating, and nondestructive testing and cleaning are appropriately controlled. Implementation of these procedures also ensures that the process is qualified to written process sheets, procedures, and checklists prepared in accordance with applicable codes, standards, and specifications. Personnel are qualified to the process through performance and verification of the successful completion of the processes. 72

All equipment, tools, materials, and testing devices used in special processes must satisfy requirements of applicable codes, standards, and specifications. Qualification records of procedures, equipment, and personnel are maintained by the station staff to ensure that those qualifications are kept current and that nonqualified personnel or equipment or out-of-date procedures are not inadvertently used.

Quality assurance personnel audit the controlling procedures for special processes and inclusion of proper references and criteria which are in compliance with applicable codes, standards, specifications, and other special requirements. When necessary, QA personnel shall perform or contract for the performance of the independent quality verification of the performance of the special processes.

Quality assurance personnel audit records generated during special processing activities to ensure that the qualifications of procedures, equipment, and personnel are current and comply with the latest revisions of criteria. Performance of these processes shall be documented and vertified by QA personnel, which will ensure compliance with all requirements of a qualified process. 15

17.2.10 Inspection (Criterion X)

The QA program includes the requirement that administrative procedures be established for inspection of activities affecting the nuclear quality-related functions of structures, systems, and components. These administrative control procedures establish the necessary requirements that assure indirect control by monitoring processing methods, equipment, and personnel. The inspection program shall be executed by or for the organization performing the activity to verify conformance with applicable administrative procedures, documented instructions, procedures, and drawings. 82

Inspections, examinations, measurements, or tests of material, products, or activities shall be performed for each work operation where necessary to ensure quality. The procedures shall identify the method and extent of inspections and should be commensurate with the importance of the activity to plant safety and reliability. Such inspections shall be performed by individuals other than those who performed or directly supervised the activity being inspected. Inspection of work

functions associated with normal station operation, routine maintenance, and certain technical services are conducted by QA/QC. Interim examinations, checks, and inspections may be accomplished by the individual performing the work. | 82

When station modification and/or major maintenance tasks are performed, inspections shall be conducted in a manner similar to that associated with original design and construction activitie.

Inspections of safety-related activities shall be performed in accordance with approved written procedures which set forth the requirements and acceptance limits and specify the inspection responsibilities.

QA shall be responsible for review of station administrative procedures and other plant documents to ensure that adequate inspection plans have been included to ensure quality. QA is further responsible for ensuring that the information concerning inspection has been obtained from related drawings, specifications, or other controlled documents. When inspection techniques require specialized qualifications or skills, personnel performing the inspection shall meet applicable licensing requirements, codes, and standards appropriate to the discipline involved. The inspection procedures shall contain a description of objectives, acceptance criteria, prerequisites, and any special conditions used to simulate normal or abnormal operating conditions as well as limiting conditions and special equipment or calibrations necessary or required to conduct the inspection, and the inspection procedure. Results of inspections shall be documented and evaluated by authorized personnel, and records shall also identify hold points if they are required to permit witnessing and shall require authorized approval to continue work beyond the hold points. Documentation of those personnel performing the inspection dates; as found conditions; deficiencies, if any; corrective action, if taken; and as left condition shall be included with the inspection records. | 72 | 72 | 15

Inspections shall be performed by personnel who have been certified by the Manager - Quality Assurance to be qualified to perform the assigned work function. It is the intent to comply with applicable quality-related requirements of Regulatory Guide 1.58 and ANSI N45.2.6-1978 in establishing the qualifications for inspection personnel. For the specific case of nondestructive examination personnel, Regulatory Guide 1.58 shall govern. The provisions of the Guide and the ANSI standard shall be reviewed, and where possible and practical, commensurate with the specific activity performed, documented in inspection activity procedures. Management measurement of each individual's qualifications shall be performed initially and at periodic intervals not to exceed 2 years to assure that the capabilities of the individual are maintained in accordance with the specified requirements or current revisions thereto. | 72 | 82

17.2.11 Test Control (Criterion XI)

A test program for ZPS-1 shall be established to ensure that structures, systems and components will perform satisfactorily in service, that the program is identified and documented, and that the testing is performed in accordance with approved written test procedures which incorporate or reference the requirements and acceptance criteria contained in applicable design documents. The test program shall include:

a. Tests during the preoperational phase to demonstrate that performance of station systems, subsystems, and integrated systems is in accordance with design intent.

b. Tests during the initial startup and operational phase to demonstrate that performance of systems and integrated plant operation that could not be tested prior to operation, is satisfactory. Additionally, confirmation of physical parameters, hydraulic or mechanical characteristics that need to be known shall be documented.

c. Surveillance tests during the operational phase to determine continuing operational readiness and adequacy of those systems and components which are normally in a standby condition, or to evaluate whether there has been any degradation of performance or any departure from the prescribed operating conditions for the systems or components normally in service. 15

d. Tests during design, fabrication, and installation activities associated with station modification and/or maintenance and tests to demonstrate satisfactory performance following maintenance, modification, or procedural change.

All types of tests are identified, documented, and controlled in accordance with written administrative procedures. Each test is accomplished and documented in accordance with written controlled test procedures.

The administrative procedures controlling the test program identify the necessary test procedures, the provisions to be included in those procedures, and the method of reviewing and approving those procedures. Test results are documented, analyzed, and evlauated by the individual or group that prepared the test procedure to ensure that these requirements are satisfied. Nuclear quality-related test results are then reviewed by the station review board, which recommends approval or disapproval by the Plant Manager. If test results are not compatible with acceptance criteria, the responsible individual or group reviews the test results, makes appropriate disposition, and initiates corrective action in accordance with corrective measures and procedures stated in the quality assurance manual. 72 82

e. If any calibration, testing, or measuring device is consistently found to be out of calibration, it shall be repaired or replaced.

f. The complete status of all items under the calibration system shall be recorded and maintained.

g. Reference and transfer standards shall be traceable to nationally recognized standards and shall be documented and retrievable. Additionally, the procedure requires that documented evidence be available to assure that the reference standard has been appropriately selected for its intended use. Requirements for reference standards will specify an uncertainty requirement of 1/4 of the tolerance of the plant process equipment being calibrated. When greater uncertainty exists, documentation will be available to explain the limitation. Where national standards do not exist, provision shall be established to document the basis for calibration. | 82 | 82

The Plant Manager is responsible for implementing this program through the station staff. QA ensures the use of approved written procedures in controlling the calibration and use of all measuring and test equipment. | 72

QA shall be responsible for reviewing and approving the administrative procedures for control of measuring and test equipment to verify the incorporation of the above specified QA requirements. Periodic audits shall be conducted by QA to verify the proper use of procedures and assess the adequacy of the measures that have been implemented to control measuring and test equipment used at ZPS-1. | 72 15 | 72

17.2.13 Handling, Storage and Shipping (Criterion XIII)

The QA program establishes measures to control the handling, storage, shipping, cleaning, and preservation of material and equipment in accordance with established instructions, procedures, or drawings. These measures shall include provisions which ensure that:

a. For particular items, special coverings, special equipment and/or special environmental considerations shall be specified, provided, and their existence verified.

b. For critical, sensitive, perishable, or high value items, specific written procedures for handling, storing, packaging, shppping, and preserving should be used. Special handling tools and/or equipment should be provided and controlled to ensure safe and adequate handling.

c. Special handling tools and equipment are inspected and tested in accordance with written procedures to verify that they are adequately maintained.

QA shall be responsible for review and approval of the records management administrative procedures to ensure that all applicable codes, standards, retentive periods, types of records, etc., are adequately addressed. The station document center shall be the final repository for all QA records associated with the design, construction, and operation of the plant. Certain records shall be maintained by offsite vendors, suppliers, contractors, AE's, etc., as provided by the QA Manual for design and construction The station document center is located in the service section of the plant and is designed, constructed, and secured to prevent inadvertent record destruction or damage by fire, flood, or deterioration by adverse environmental conditions such as temperature or humidity. 72

17.2.18 Audits (Criterion XVIII)

A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the QA program and the controlling administrative procedures. Audits of selected aspects of operating phase activities shall be performed with a frequency commensurate with their safety significance and in such a manner as to ensure that an audit of all safety-related activities is completed within a 2-year period. The review and audit program is described in Subsection 17.2.2.5.

The Manager - QA will be responsible for directing planned and documented audits to verify compliance with all aspects of the QA program. These audits should verify compliance with and effectiveness of implementation of internal rules, procedures, regulations and license provisions; programs for training; and corrective actions and record management. 72 15

Additionally, audits of vendors', suppliers', or contractors' QA programs and records are performed under the direction and supervision of the Manager - QA to ensure that nuclear quality-related services, activities or supplied equipment, parts, or components comply with established quality criteria. 72

Internal audits of station activities should be periodically performed by members of the station staff. These internal audits shall be the responsibility of the Plant Manager and should provide a continuing evaluation and documented history of plant operation under the jurisdiction of the Plant Manager. 72 72

TABLE 17.2-1

STATION ACTIVITIES LIST

1 - Preoperational Testing
2 - Startup Testing
3 - Surveillance Testing
4 - Maintenance; corrective and preventive
5 - Instrument calibration and test
6 - Operations
7 - Radwaste Management
8 - Radiation Protection
9 - Chemistry and radiochemistry
10 - Emergency Planning
11 - Materials Control
12 - Inspection and audit
13 - Training
14 - Security
15 - Document Control
16 - Records Management
17 - Nuclear Engineering

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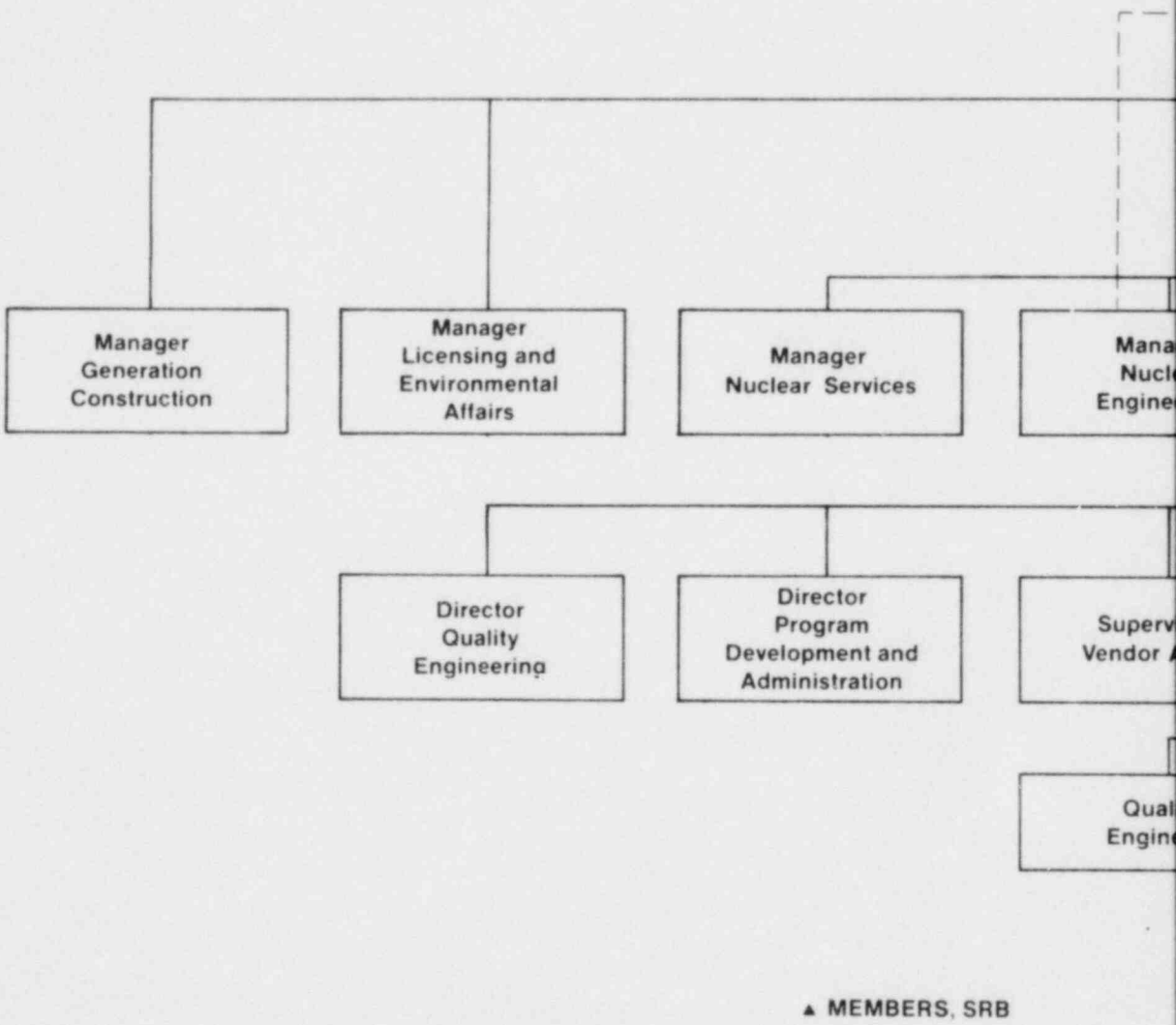
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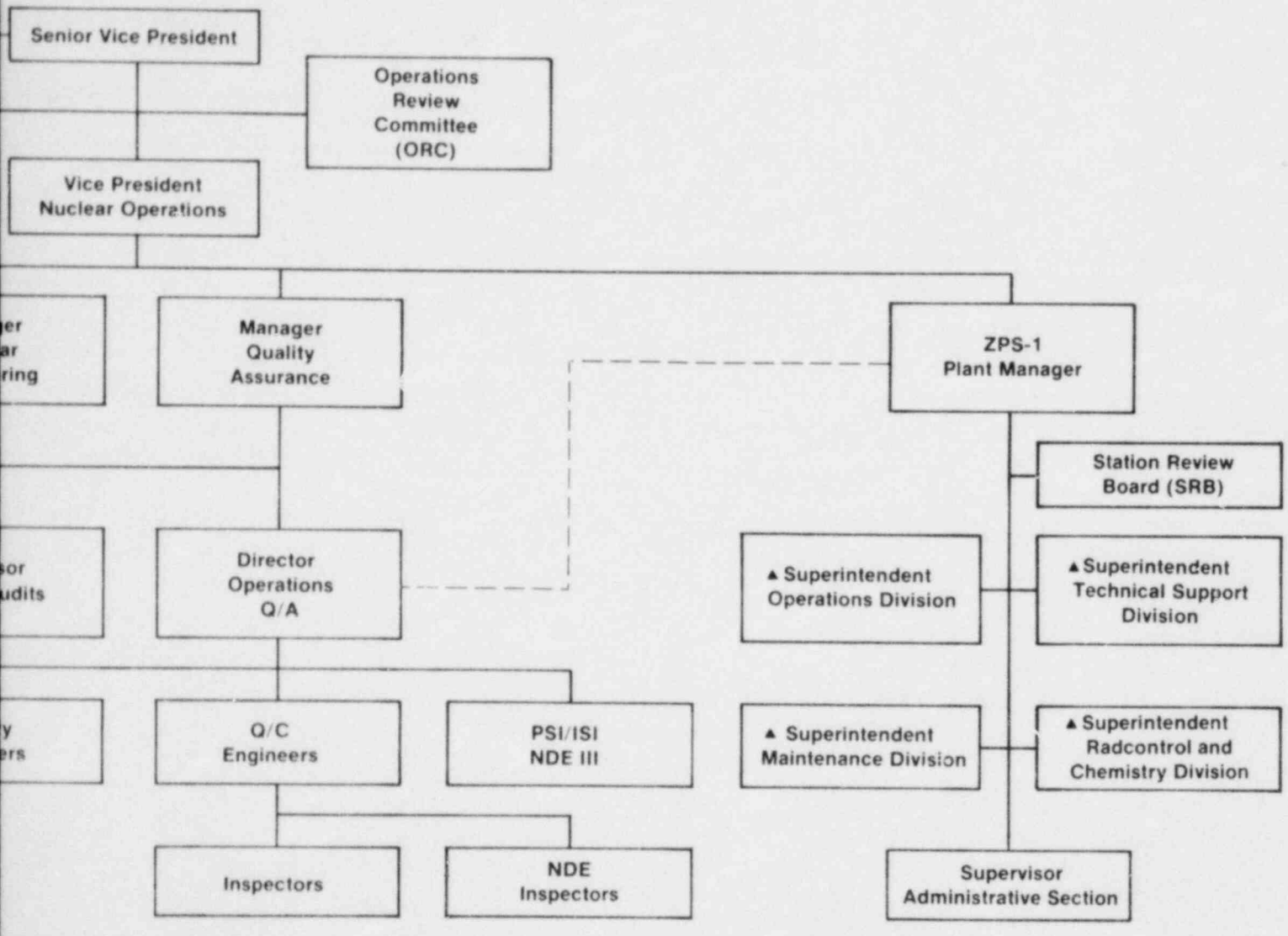

WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1
FINAL SAFETY ANALYSIS REPORT

FIGURE 17.2-1

CG&E ORGANIZATION FOR QUALITY ASSURANCE

REGULATORY GUIDE 1.7 (REV. 1)

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Applicant's Position:

The applicant meets the intent of this regulatory guide. The station is equipped with a hydrogen-oxygen recombiner to prevent the hydrogen concentration from reaching a flammable limit within the primary containment. Per the requirements of this Regulatory Guide, provisions have been included to allow the standby gas treatment system to be used to aid in purging the primary containment during postaccident cleanup. The applicant has also made a commitment to inert the containment. For details, see Subsection 6.2.5 and Appendix L, Section L.28. 82

REGULATORY GUIDE 1.66 (REV. 0)

NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS

Applicant's Position:

The Wm. H. Zimmer Nuclear Power Station Unit 1 meets the intent of Regulatory Guide 1.66 (Revision 0). 82

POSITION 130.31 | 82

"It is stated in Subsection 3.7.2.1.2.7.1 of the FSAR, 'When the piping system is being anchored and supported at points with different excitations, the response spectrum analysis is performed using the response spectrum above the center of mass of the piping system.'

"This approach is inconsistent with the staff's position delineated in Section 3.7.3 of the SRP. The staff's position requires that an envelope of all pertinent response spectra be used for the analysis of systems spanning between two or more supports.

"Please provide technical justification for the deviation and demonstrate that your approch would result in an equivalent or more conservative analysis than is required. The justification may consist of a comparison of inputs, a comparison of selective results, or a combination of both."

RESPONSE

Subsection 3.7.2.1.2.7.1 of the FSAR, as presently stated, was applicable to the piping analysis when seismic loading was the only dynamic loading input to the analysis. With the addition of SRV and LOCA loads to the design basis, the response spectrum analysis methodology was revised to provide an envelope of the response spectra for the analysis of piping spanning two or more supports. | 72

Piping analysis on the Zimmer Plant is conservative and does comply with the requirements of Section 3.7.3 of the SRP. Subsection 3.7.2.1.2.7.1 of the FSAR will be revised to reflect the current methods of analysis. Subsection 3.7.2.1.2.7.2 on equipment is also revised. | 82

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h = barrier thickness, ft,

K_P = penetration coefficient in modified Petry formula, ft^3/lb, and

W/A = weight of missile per unit area (lb/ft^2).

The values of K_P used in this analysis are 0.0038 ft^3/lb for structural concrete walls and 0.0048 ft^3/lb for block walls.

If the missile is not able to perforate the wall, its rebound velocity is calculated using the energy balance approach (Reference 2).

e. The missile proceeds in the same direction following the barrier perforation. The direction of the missile following a ricochet is calculated using the incident velocity and angle.

f. In tracking the path of the missile, note any targets hit by the missile. No credit is taken for the target reducing the missile velocity, nor is credit taken for probability that the missile impact may not disable the target (component).

g. The missile is terminated when the missile energy drops below a preselected minimum of 10^4 ft-lb or if it exits the auxiliary building.

In the auxiliary building between elevations 546 feet and 0 inches and 591 feet 7 inches, seven vital targets are identified. these are marked 1101, 1102, 1201, 1202, 1203, 1204, and 1205. Target 1101 consists of control room panels of both the safety divisions, A and B. Target 1201 consists of HVAC equipment of both safety divisions. These targets (1101 and 1201) are therefore labeled C. Target 1102 is the division A battery room. Targets 1202 and 1203 are electric cable pans feeding to safety Division A, and targets 1204 and 1205 are electric cable pans of safety division B. Targets 1102, 1202, and 1203 are therefore labeled A, and targets 1204 and 1205 are marked B.

In 1000 trials (disc breakups), no C targets were hit by the missiles. There were 20 hits on target 1102. There were no simultaneous hits on the targets labeled A and B. There were therefore no safety compromises from turbine missiles in the auxiliary buiilding. The probability of safety compromise as such from damage to components in the auxiliary building is less than $1x10^{-3}$ per turbine failure at destructive overspeed. This probability multiplied by the turbine failure probability, P_1, of $4x10^{-5}$ per year (as suggested by the NRC) results in the overall probability of turbine missile damage of $4x10^{-8}$ per year. This overall probability, P_4, is much lower (equal to $1.7x10^{-9}$ per year) if the more appropriate value of P_1 equal to $1.7x10^{-6}$ per year calculated by Westinghouse is used.

POSITION 110.28 (5.2.1, Position 110.25)

"Verify that the 2179 KIP shear and 33,088 ft-KIP moment loads which were referenced in the response to Question 110.25 include both the internal and external pressures resulting from blowdown forces. In addition, for resultant loads due to both internal and external pressures, provide the margin of safety based on the stress intensity limits of the material for the following reactor vessel support structural elements: (1) tie-down bolts, (2) skirt and skirt flange junction, (3) skirt knuckle, and (4) skirt knuckle and bottom head junction." 46

RESPONSE

The loads given in response to Position 110.25 were based on the mass-energy blowdown flow rates for the recirculation line break as defined at that time. The blowdown has since been reevaluated and new loads calculated from the annulus pressurization.

Both internal and external pressures were considered. The method of analysis is described in the Design Assessment Report, Subsection 5.3.1.4, and FSAR Subsection 3.9.1.1.3.

The new loads acting at the RPV support skirt are listed below. The loads are SRSS values or larger (ABS values were used for the vertical loadings to give conservative results). Only the faulted condition includes annulus pressurization loads.

OPERATING CONDITION	VERTICAL, kips	SHEAR, kips	MOMENT, in.-kips x 10^{-3}
Upset	5938	403	78
Emergency	7367	403	78
Faulted	10838	2756	252

82

The resultant stress intensities and margins of safety in the tie-down bolts are given in Section 7.4 of the DAR, Amendment 17. The new values are reproduced below:

OPERATING CONDITION	TENSION, kips/bolt	SHEAR, kips/bolt	MARGIN, FACTOR
Upset	61.15	6.7	1.9
Emergency	73.00	6.7	1.6
Faulted	128.00	45.9	1.3

The region of the skirt having the lowest design margin is the knuckle-to-cylinder junction region. Values of stress intensity and factor of safety for this region may be found in FSAR Table 3.9-3 for the limiting transient for each operating condition. The margins of safety for the skirt and skirt flange junction, and for the skirt knuckle and bottom head junction are higher than those for the knuckle to skirt regions.

POSITION 310.11 (15.1) | 78 | 79

"(Section 15.1.9). Provide a conservative design basis loss of coolant dose calculation which reflects resolution of the discrepancies listed in questions 310.7 and 310.8 which assumes direct release of fission products from the primary containment to the environment during any time period that the secondary containment volumes are at less than 0.25 inch water guage below ambient pressure and which assumes mixing in only 50% of the volume between inlet and outlet headers in the secondary containment volume. Also, provide the additional dose consequences resulting from the following:

a) Operation of the main steam line isolation valve leakage control system which you will provide (See Regulatory Guide 1.96).

b) Operation of the hydrogen purge system assuming the recombiners do not function. Include the time at which the purge is initiated and the purge rate."

RESPONSE | 78 | 79

The discrepancies in Positions 310.7 and 310.8 are resolved in this revision.

a. The technical specification for main steam isolation valves will be established after initial testing.

   Typical doses of the EAB and LPZ are presented in Table 9.3-4 for several leak rates.

b. The hydrogen recombiner is redundant in all active components and meets the requirements of Regulatory Guide 1.7 and is designed as an engineered safeguard system. Nitrogen inerting of the containment has been added. Provisions have been included to allow the standby gas treatment system to be used to aid in primary containment postaccident cleanup. | 79 | 79 | 8

POSITION 120.5 (APPENDIX C) |77|78|82

"Indicate conformance with or provide justification for not following the recommendations of the following Regulatory Guides: 1.31, 1.34, 1.43, 1.44, 1.45, 1.50, 1.51, 1.56, 1.65, 1.66, and 1.71." |78

RESPONSE

1. Regulatory Guide 1.51

   This guide has been withdrawn.

2. Regulatory Guides 1.50 (Revision 0) and 1.71 |77

   Prior to beginning any welding operations, the project welding engineer or his designee reviews all pertinent drawings and documents prior to the issuance of a weld card. The engineer then prepares a packet for each weld which identifies qualified procedures to be used, preheat, interpass, and postheat requirements, NDE requirements, and special procedures to be followed. Limited accessibility is a specific area of concern in the welding engineer's review plan (Regulatory Guide 1.71). Control of preheat temperatures for welding of low-alloy steel is also a specific concern in the welding engineer's review plan (Regulatory Guide 1.50, Revision 0). |78|82 |77

   The QA engineer prepares an inspection plan for the specific weld, and this plan is also included in the packet kept at the weld site.

   Appropriate signoffs are required to ensure that all requirements are met. The QA engineer monitors the welding operations and the packet signoffs. |77

3. Regulatory Guides 1.31, 1.34, 1.43, 1.44, 1.45 (Revision 0), 1.56 (Revision 0), 1.65, and 1.66 |78

   Information on conformance with these guides is presented in Appendix C of the FSAR. |29

L.72 APPROACH USED TO INERT CONTAINMENT (II.B.7 and .8) Status of SER NUREG-0737 Items

78

The accident at Three Mile Island, Unit 2, resulted in a severely damaged core accompanied by the generation and release to containment of hydrogen in excess of those limits considered in current regulations. This accident highlighted the difficulties associated with mitigating the consequences of accidents more severe than the current design-basis accidents. As a result, the TMI Action Plan (NUREG-0660), item II.B.8, called for a rulemaking proceeding for consideration of degraded or melted cores in safety reviews. Additionally, the TMI Action Plan, item II.B.7, discussed analysis of hydrogen control and the need for inerting small containments. Both of these requirements for addressing these items are in NUREG-0694.

Subsequent to the issuance of TMI Action Plan items II.B.7 and II.B.8, 10 CFR 50.44 was revised and now requires that an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment.

Per 10 CFR 50.44, Zimmer Power Station has a system design for nitrogen inerting. Engineering has been completed. Procurement of equipment has begun and is expected to be completed by May 15, 1982. The schedule calls for complete installation by August 15, 1982.

82

The primary containment inerting system is designed to maintain the drywell and suppression chamber atmosphere at less than 4% oxygen during reactor power operation. This condition, as specified in the station Technical Specifications, must be met and maintained within the 24-hour period after achieving greater than 15% of rated thermal power, or 365 MWt. Deinerting the primary containment may commence 24-hours prior to reducing reactor power to less than 15% of rated thermal power preliminary to a scheduled reactor shutdown.

The inerting system is designed to provide gaseous nitrogen into the dyrwell and suppression chamber at a flow rate of up to 150,000 SCFH prior to power operation as described in the preceding paragraphs. A feed and bleed operation directs the nitrogen into the containment, as shown on Figures 6.2-64 and 9.4-7. Primary containment purge, or effluent, is routed through the primary containment and suppression pool purge (VQ) system. During normal reactor operation, gaseous nitrogen is provided, as required, to maintain oxygen concentrations at the Technical Specification requirement of less than 4% oxygen by volume.

For initial inerting requirements, liquid nitrogen is routed from the storage tank farm to a vaporizer system to a high flow/temperature control station and an appropriate pressure regulator set at 40 psig.

Low-flow, makeup nitrogen requirements are satisfied by routing gaseous nitrogen, at up to 5000 SCFD, from the gaseous space of the liquid nitrogen storage tank farm, through an ambient temperature vaporizer, a pressure regulator set at 40 psig, and to the primary containment.

Both high-flow and low-flow systems are designed to protect downstream piping from liquid nitrogen by a low temperature shutoff valve which trips and isolates flow upon actuation of a temperature control circuit.

As noted in the preceding paragraphs, containment effluent is routed through the containment purge (VQ) package filters. Each package filter unit consists of a prefilter, HEPA filter, charcoal bed, HEPA filter, exhaust fan, and appropriate instrumentation, control, and fire protection. The use of these filter units ensures that the SBGTS is preserved for use during postulated accident situations and prevents their degradation due to normal plant operation. Safety-related motor-operated isolation valves meet containment isolation criteria, and duct work and piping, up to secondary containment isolation, are designed to Seismic Category I criteria. 82

10 CFR 50.8 - REPORTING, RECORDKEEPING, AND APPLICATION REQUIREMENTS: OMB APPROVAL

STATEMENT OF SECTION 50.8

(a) the Nuclear Regulatory Commission has submitted the information collection requirements contained in this part of the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act (Pub. L. 96-511). OMB approved the information collection requirements on October 30, 1981.

(1) The OMB approval number is 3150-0011.

(2) OMB approval expires April 30, 1982.

(b) The approved information collection requirements include the application, recordkeeping, and reporting requirements contained in Sections 50.30; 50.33; 50.33a; 50.34(b), (c), (d), (f); 50.34a; 50.35(b); 50.36; 50.36a; 50.48; 50.54(f), (p), (q), (r), (s), (t), (u); 50.55(e); 50.55a; 50.59(b), (c); 50.71(a), (b), (c), (d), (e); 50.72(a), (b); 50.80; 50.82; 50.90; and Appendices A, B, C, E, G, H, J, K, and R. 82

EFFECTIVE DATE

Section 50.8, as added, effective December 30, 1981 (46 F.R. 63033).

EVALUATION OF COMPLIANCE

This regulation imposes no requirements on the licensee.

CONCLUSION

No requirements.

STATEMENT OF SECTION 50.34(e)

Each applicant for a license to operate a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of Section 73.21 of this chapter, as appropriate. 82

EVALUATION AND CONCLUSION

ZPS-1 complies with this section.

STATEMENT OF SECTION 50.34(f)

Additional TMI-related equipments. In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982 shall meet the requirements in paragraphs (b)(1) through (3) of this section. This rule applies only to the pending applications by Duke Power Company (Perkins Nuclear Station Units 1, 2 and 3), Houston Lighting & Power Company (Allens Creek Nuclear Generating Station, Unit 1), Portland General Electric Company (Pebble Springs Nuclear Plant, Units 1 and 2), Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Puget Sound Power & Light Company (Skagit/Hanford Nuclear Power Project, Units 1 and 2), and Offshore Power Systems (License to Manufacture Floating Nuclear Plants. The number of units that will be specified in the manufacturing license, if issued, will be that number whose start of manufacture, as defined in the license application, can practically begin within a ten-year period commencing on the date of issuance of the manufacturing license, but in no event will that number be in excess of ten. The manufacturing license will require the plant design to be updated no later than five years after its approval. Paragraphs (b)(1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this rule. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and Criterion 50 of Appendix A to 10 CFR Part 50, is sufficient for issuance of the manufacturing license may be considered in the manufacturing license proceeding. 82

(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license.[6]

[6] Alphanumeric designations correspond to the related action plan items in NUREG-0718 and NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." They are provided herein for information only.

(i) Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)

(ii) Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only) (II.B.1.1):

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(B) A design review of AFWS.

(C) An evaluation of AFW flow design bases and criteria.

(iii) Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

(iv) Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure fails after the PORV has opened. (Applicable to PWR's only). (II.K.3.2) 82

(v) Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.13)

(vi) Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)

(vii) Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

(viii) Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.24)

(x) Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28) 82

(xi) Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)

(xii) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (b)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (b)(2)(ix) of this section.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.

(i) Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only) (I.A.4.2)

(ii) Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)

(iii) Provide, for Commission review a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)

(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2) 82

(v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)

(vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases chloride, and boron concentrations. (II.B.3)

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8).

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system. 82

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing not be carried cut until subsequent phases of the test program are developed. (II.D.1)

(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)

(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)

(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)

(xiv) Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4) 82

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)

(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)

(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)

(xxii) Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B & W-designed plants only). (II.K.2.9)

(xxii) Provide as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B & W-designed plants only). (II.K.2.10)

(xxiv) Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23) 82

(xxv) Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2)

(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)

(xxvii) Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)

(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. (III.D.3.4)

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.

(i) Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)

(ii) Ensure that the quality assurance (QA) list required by Criterion II, App. B, 10 CFR Part 50 includes all structures, systems, and components important to safety. (I.F.1)

(iii) Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functioning at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities. (I.F.2) 82

(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

(v) Provide preliminary design information at a level of detail consistent with the normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)

(A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements

set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraph (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H St., NW., Washington, D.C.

(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category, (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting. 82

(vi) For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)

(vii) Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architent engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the

transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

EFFECTIVE DATE

Amended January 15, 1982; effective February 16, 1982 (47 F.R. 2286). 82

EVALUATION OF COMPLIANCE

Not applicable to ZPS-1.

10 CFR 50.35 - ISSUANCE OF CONSTRUCTION PERMITS

STATEMENT OF SECTION 50.35

Issuance of construction permits.[2] (a) When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features, the Commission may issue a construction permit if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public; (2) such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report; (3) safety features of components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that (4) on the basis of the foregoing, there is reasonable assurance that, (i) such safety questions will be satisfactorily resolved at or before the latest date and stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in Part 100 of this chapter, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public. 77

NOTE: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the facility, the findings required above will be appropriately modified to reflect that fact.

(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at his option, may request such approvals in the construction permit or, from time to time, by amendment of his construction permit. The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results

[2]The Commission may issue a provisional construction permit pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing of an applicant for a provisional construction permit has been published on or before that date.

of research and development programs designed to resolve safety questions.

(c) Any construction permit will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until (1) the applicant has submitted to the Commission, by amendment to the application, the complete safety analysis report, portions of which may be submitted and evaluated from time to time, and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and the regulations in this chapter.

EFFECTIVE DATE

Amended April 17, 1970; effective April 30, 1970 (35 F.R. 6644)

EVALUATION OF COMPLIANCE

Most of the section does not apply since ZPS-1 already has a construction permit. CG&E has submitted a complete Safety Analysis Report in the FSAR as a part of its application for an operating license. 77

CONCLUSION

ZPS-1 complies with the applicable portions of this section.

6.2.5.2.2). Sampling point locations are described in the response to Question 041.33; system description is provided in Subsections 7.6.1.8.4 and 7.6.2.8.

(2) As described in FSAR Subsection 6.2.5.2.1 on the response to Question 041.33, containment atmosphere mixing is ensured through natural convection, turbulence associated with containment sprays and the steam escaping from the LOCA event, and airflow patterns created by the hydrogen recombiner intake and discharge locations.

(3) Hydrogen generation analyses were performed for ZPS-1 for the time following a postulated LOCA event. The results are presented in FSAR Subsection 6.2.5.3 and the response to Position 042.7. The ZPS-1 hydrogen control systems maintain the hydrogen concentration in containment below 4% by volume. 77

STATEMENT OF SECTION 50.44(c)

(1) For each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets within cylindrical zircaloy cladding, it shall be shown that during the time period following a postulated LOCA but prior to effective operation of the combustible gas control system, either: (i) An uncontrolled hydrogen-oxygen recombination would not take place in the containment; or (ii) the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function. 82 82 82

(2) If the conditions set out in paragraph (c)(1) of this section cannot be shown, the containment shall be provided with an inerted or an oxygen deficient atmosphere in order to provide protection against hydrogen burning and explosions during the time period specified in paragraph (c)(1) of this section.

(3) Notwithstanding paragraphs (c)(1) and (c)(2) of this section:

(i) Effective May 4, 1982 or 6 months after initial criticality, whichever is later, an inerted atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment; and

(ii) By the end of the first scheduled outage beginning after July 5, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor that relies upon a purge/repressurization system as the primary means of controlling combustible gases following a LOCA shall be provided with either an internal recombiner or the capability to install an external recombiner following the start of an accident. The internal or external recombiners must meet the combustible gas control requirements in paragraph (d) of this section. The containment penetrations used for externals recombiners must either be: 82

(A) dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, be designed against postulated single failures for containment isolation purposes, and be sized to satisfy the flow requirements of the external recombiners, or

(B) of a combined design for use by either external recombiners or purge/repressurization systems and other systems, conform to the requirements of criteria 54 and 56 of Appendix A of this part, be designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/repressurization systems, and be sized to satisfy the flow requirements of the external recombiners or purge repressurization systems.

(iii) To provide improved operational capability to maintain adequate core cooling following an accident, by the end of the first scheduled outage beginning after July 1, 1982 and of sufficient duration to permit required modifications, each light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. (High point vents are not required, however, for the tubes in U-tube steam generators.) The high point vents must be remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B of this part. In particular, the vent system shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident. 82

EVALUATION OF COMPLIANCE

(1) The hydrogen generation calculation presented in FSAR Subsection 6.2.5.3 indicates that the hydrogen concentration in the primary containment will not reach 4% by volume until 16 hours after a LOCA event. The reactor operator will begin warming up the recombiner 30 minutes after a LOCA and will initiate recombination 3.5 hours after the LOCA (FSAR Subsection 6.2.5.2.3.2); operation of the recombiner prevents the hydrogen concentration from ever reaching the 4% by volume level. The adequacy of the 4% value is documented in Regulatory Guide 1.7, Revision 2, November 1978: "... the NRC staff concludes that a lower flammability limit of 4 volume percent hydrogen in air or steam-air atmospheres is well established and is adequately conservative." Therefore, an uncontrolled hydrogen-oxygen 77

recombination would not take place in the containment and inerting would not be required.

(2) Despite the above, CG&E is installing a nitrogen inerting system in ZPS-1 which will be operational within the time frame specified in Section 50.44(c)(1). 77 82

(3)(i) The Zimmer Power Station has a system design for nitrogen inerting. Engineering has been completed. Procurement of equipment has begun and is expected to be completed by May 15, 1982. The schedule calls for complete installation by August 15, 1982.

(3)(ii) The Zimmer Power Station has been provided with a permanently installed, dedicated recombiner for controlling combustible gases following a LOCA. This internal recombiner meets the combustible gas control requirements in paragraph (d) of this section. 82

(3)(iii) High point vents are provided for the reactor coolant system, for the vessel head, and for other systems required to maintain adequate core cooling. These vents are provided in order to prevent loss of function of these systems due to the accumulation of noncondensible gases. The high point vents are remotely operated from the control room. The design of the vents, associated controls, instruments, and power sources meet the intent of Appendices A and B of this part. The requirements of this section are more completely addressed in a thorough discussion of this system as provided in Appendix L to the FSAR under Item L.19

STATEMENT OF SECTION 50.44(d)

(1) For facilities that are in compliance with Section 50.46(b), the amount of hydrogen contributed by core metal-water reaction (percentage of fuel cladding that reacts with water), as a result of degradation, but not total failure, of emergency core cooling functioning shall be assumed either to be five times the total amount of hydrogen calculated in demonstrating compliance with Section 50.46(b)(3), or to be the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch (0.0058mm), whichever amount is greater. A time period of 2 minutes shall be used as the interval after the postulated LOCA over which the metal-water reaction occurs. (2) For facilities as to which no evaluation of compliance in accordance with Section 50.46(b) has been submitted and evaluated, the amounts of hydrogen so contributed shall be assumed to be that amount resulting from the reaction of 5 percent of the mass of metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume. 77

EVALUATION OF COMPLIANCE

The ZPS-1 hydrogen concentration curves given in the response to Position 042.7 are based on a metal-water reaction of cladding to a depth of 0.00023 inch.

STATEMENT OF SECTION 50.44(e)

For facilities whose notice of hearing on the application for a construction permit was published on or after November 5, 1970, purging and/or repressurization shall not be the primary means for controlling combustible gases following a LOCA. However, the capability for controlled purging shall be provided. For these facilities, the primary means for controlling combustible gases following a LOCA shall consist of a combustible gas control system, such as recombiners, that does not result in a significant release from containment.

EVALUATION OF COMPLIANCE

The ZPS-1 notice of hearing on the application for a construction permit was issued on March 7, 1972 (37 F.R. 4925). As described in the evaluations of other paragraphs of this section, ZPS-1 uses recombiners as the primary means of controlling combustible gases. The capability for controlled purging is provided. 77

STATEMENT OF SECTION 50.44(f)

For facilities with respect to which the notice of hearing on the application for a construction permit was published between December 22, 1968, and November 5, 1970, if the incremental radiation dose from purging (and repressurization, if a repressurization system is provided) occurring at all points beyond the exclusion area boundary after a postulated LOCA calculated in accordance with Section 100.11(a)(2) of this chapter is less than 2.5 rem to the whole body and less than 30 rem to the thyroid, and if the combined radiation dose at the low population zone outer boundary from purging and the postulated LOCA calculated in accordance with Section 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, the purge system shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure.

EVALUATION OF COMPLIANCE

This paragraph is not applicable because the ZPS-1 notice of hearing was issued March 7, 1972 (37 F.R. 4925).

STATEMENT OF SECTION 50.44(g)

For facilities with respect to which the notice of hearing on the application for a construction permit was published on or before December 22, 1968, if the combined radiation dose at the low population zone outer boundary from purging (and repressurization, if a repressurization system is provided) and the postulated LOCA calculated in accordance with Section 100.11(a)(2) of this chapter is less than 25 rem to the whole body and less than 300 rem to the thyroid, only a purging system is necessary, provided that the purging system and any filtration system associated with it are designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. Otherwise, the facility shall be provided with another type of combustible gas control system (a repressurization system is acceptable) designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. If a purge system is used as part of the repressurization system, it shall be designed to conform with the general requirements of Criteria 41, 42, and 43 of Appendix A to this part. The containment shall not be repressurized beyond 50 percent of the containment design pressure. 77

EVALUATION OF COMPLIANCE

This paragraph is not applicable to ZPS-1 because the notice of hearing on the application for a construction permit was published March 7, 1972 (37 F.R. 4925).

SUMMARY OF SECTION 50.44(h)

This section provides definitions for the degradation of the ECCS, combustible gas control system, purging system, and repressurization system.

EVALUATION OF COMPLIANCE

This paragraph places no requirements on the licensee.

CONCLUSION

ZPS-1 complies with this section.

REFERENCES

1. FSAR Subsection 6.2.5.

2. FSAR Subsection 7.6.1.8.4.

3. FSAR Subsection 7.6.2.8.

4. Response to Position 042.7.

5. Response to Position 041.33.

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10 CFR 50.47 - EMERGENCY PLANS

STATEMENT OF SECTION 50.47(a)

(1) No operating license for a nuclear power reactor will be issued unless a finding is made by NRC that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

(2) The NRC will base its finding on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on the NRC assessment as to whether the applicant's onsite emergency plans are adequate and capable of being implemented. In any NRC licensing proceeding, a FEMA finding will constitute a rebuttable presumption on a question of adequacy.

EFFECTIVE DATE

Section 50.47 was added August 19, 1980; effective November 3, 1980 (45 F.R. 55402).

EVALUATION OF COMPLIANCE

These paragraphs impose requirements on the NRC; they do not impose any requirements on the licensee. 77

STATEMENT OF SECTION 50.47(b) 82

The onsite and offsite emergency response plans for nuclear power reactors must meet the following standards:[1]

(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is

[1]These standards are addressed by specific criteria in NUREG-0654; FEMA-REP-1 entitled, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants - for Interim Use and Comment," January 1980.

(11) Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

(12) Arrangements are made for medical services for contaminated injured individuals.

(13) General plans for recovery and reentry are developed.

(14) Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

(15) Radiological emergency response training is provided to those who may be called on to assist in an emergency.

(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

EVALUATION OF COMPLIANCE 77

CG&E submitted an Emergency Plan for ZPS-1 in Appendix F to the FSAR; it was written to comply with the regulations and NRC guidance in effect at the time of its submittal (September 10, 1975). As a result of the TMI accident, the NRC issued this newly revised regulation and other documents to upgrade emergency planning. In response to the new requirements, CG&E redrafted the Emergency Response Plan and submitted the draft to the NRC for review on January 30, 1981 and on July 1, 1981. Upon receiving NRC approval for the new plan, and prior to issuance of the ZPS-1 operating license, the revised plan will be included in the FSAR. The following analysis is based on the most recent draft of the Emergency Response Plan. 82

(1) Responsibilities for emergency response are provided in Sections F.5 and F.6 of the plan. CG&E is responsible to provide State and local organizations with information and recommendations concerning the emergency situation. Offsite organizations respond as outlined in their respective state and county plans (included in the ZPS-1 plan). Augmentation of CG&E's onsite emergency organization is described in Subsection F.5.3; staffing requirements for offsite organization are addressed in their respective plans.

(2) Licensee actions and responsibilities are clearly defined in Sections F.4 and F.6 and Appendix A of the Emergency Response Plan. Discussion of staffing requirements and duties for initial accident response is included in Subsection F.5.2 and Table

F.5.1. Timely augmentation of the onsite emergency organization is available as discussed in Subsection F.5.3. Interfaces between onsite and offsite emergency response organizations are detailed in Subsections F.5.3.2 and F.5.4; Table F.5-3; Subsections F.6.1.2 and F.6.4.1.2; and Appendices D, E, F, and G.

(3) Arrangements for obtaining assistance resources are described in Emergency Response Plan Subsections F.5.4 and F.9.2. As described in Subsection F.7.1.6, the Emergency Operations Facility is sized to accommodate at least 35 people; this arrangement will allow the facility to accommodate state and local emergency response staff. Organizations included in the planned response are described in Subsections F.5.3.2 and F.5.4.

(4) A standard emergency classification and action level scheme, based on factors including facility system and effluent parameters, is used as described in Emergency Response Plan Section F.4. The classification scheme is based on Appendix 1 to NUREG-0654/FEMA-REP-1 (Revision 1), "Emergency Action Level Guidelines for Nuclear Power Plants," November 1980. State and county response plans (included as Appendices E, F, and G to the Emergency Response Plan) require CG&E to provide information to determine initial offsite response measures. Information provided by CG&E is listed in Subsection F.6.1.2. 82

(5) Procedures have been established for notification of State and local response organizations; Emergency Response Plan Sections F.5 and F.6 and Appendix A describe the notification procedure and the content of initial and follow-up messages to the state and local organizations. Responsibilities of offsite organizations are described in their respective state and county emergency plans. Means for public notification and instruction are described in Emergency Response Plan Subsection F.6.4.1.2.2 and in the State and local emergency plans. 77

(6) Means for prompt communications among response organizations and emergency personnel and the public have been established as described in the Emergency Response Plan Sections F.5 and F.6 and the State and local emergency plans.

(7) As described in Emergency Response Plan Subsection F.8.5, CG&E provides for periodic dissemination of information to the public regarding how they will be notified and what their actions should be during a radiological emergency. Warning or advising the public of emergency actions is the responsibility of State and local response organizations in accord with their emergency plans. Coordinated dissemination of information to the public is handled by the Public Information Manager, as described in Appendix A to the Emergency Response Plan.

(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained as detailed in Emergency Response Plan Section F.7.

(9) Onsite and offsite methods, systems, and equipment for assessing and monitoring offsite consequences of a radiological emergency are provided (Emergency Response Plan Section F.7).

(10) A range of protective actions has been developed and reported in Subsection F.6.4 of the Emergency Response Plan. Actions are included for onsite activities for the EPZ, and for the ingestion exposure pathway EPZ.

(11) Control of emergency response personnel exposure is discussed in Emrgency Response Plan Subsection F.6.5.1. In all situations, every reasonable effort is made to minimize radiation exposure of ZPS-1 emergency response personnel.

(12) Decontamination and first aid facilities are located onsite and medical transportation and treatment of injured, contaminated individuals has been provided (see Emergency Response Plan Subsection F.6.5).

(13) General plans for recovery and reentry have been prepared and are described in Section F.9 of the Emergency Response Plan.

(14) An emergency response exercise was conducted on November 18, 1981 and will be conducted periodically thereafter. Periodic drills are also conducted. A critique of each exercise is provided by qualified Federal, State, local, and corporate evaluators; any deficiencies identified are corrected. Emergency Response Plan Subsection F.8.2.2 details CG&E's plans for conducting exercises and drills. 82 77

(15) Periodic training is given to emergency response personnel, as described in Subsection F.8.2.1 of the Emergency Response Plan.

(16) The Emergency Response Plan is reviewed and updated annually by the Station Review Board and audited by the Operations Review Committee at least every 2 years (Subsection F.8.3). 82

STATEMENT OF SECTION 50.47(c)

(1) Failure to meet the standards set forth in paragraph (b) of this subsection may result in the Commission declining to issue an Operating License; however, the applicant will have an opportunity to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operation.

(2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the

STATEMENT OF SECTION 50.54(s)

(1) Each licensee who is authorized to possess and/or operate a nuclear power reactor shall submit to NRC within 60 days of the effective date of this amendment the radiological emergency response plans of State and local government entities in the United States that are wholly or partially within a plume exposure pathway EPZ, as well as the plans of State governments wholly or partially within an ingestion pathway EPZ.[1] [2] Ten (10) copies of the above plans shall be forwarded to the Director of Nuclear Reactor Regulation with 3 copies to the Director of the appropriate NRC regional office. Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs for a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway EPZ shall focus on such actions as are appropriate to protect the food ingestion pathway. 77

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(2)(i) For operating power reactors, the licensee, State, and local emergency response plans shall be implemented by April 1, 1981, except as provided in Section IV.D.3 of Appendix E of this Part. 82

(2)(ii) If, after April 1, 1981, the NRC finds that the state of emergency preparedness does not provide reasonable assurance that appropriate protective measures can and will be taken in the event of a radiological emergency (including findings based on requirements of Appendix E, Section IV.D.3) and if the deficiencies (including deficiencies based on requirements of Appendix E, Section IV.D.3) are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate. In determining whether a shutdown or other enforcement action is appropriate, the Commission shall take into account, among other factors, whether the licensee can demonstrate to the Commission's satisfaction that the deficiencies in the plant are not significant for the plant in 82 82

[1] Emergency Planning Zones (EPZs) are discussed in NUREG-0396: EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants in Support of Light Water Nuclear Power Plants," December 1978.

[2] If the State and local emergency response plans have been previously provided to the NRC for inclusion in the facility docket, the applicant need only provide the appropriate reference to meet this requirement.

EVALUATION OF COMPLIANCE

Section F.8 of the Emergency Response Plan states that the plan is reviewed every 12 months by the Station Review Board. The program is also independently audited at least every 2 years by the Operations Review Committee. Results of these reviews are recorded and retained for a period of 5 years. Drills and exercises are also performed as described in Subsection F.8.2.2. 78

STATEMENT OF SECTION 50.54(u)

Within 60 days after the effective date of this amendment, each nuclear power reactor licensee shall submit to the NRC plans for coping with emergencies that meet standards in Section 50.47(b) and the requirements of Appendix E of this Part.

EVALUATION OF COMPLIANCE

The ZPS-1 Emergency Response Plan has been submitted to the NRC and is currently undergoing review.

CONCLUSION

ZPS-1 complies with this section notwithstanding certain exemptions to paragraph(o) which have been requested. 77

STATEMENT OF SECTION 50.54(v)

(v) Each licensee subject to the requirements of Part 73 of this chapter shall ensure that physical security, safeguards contingency and guard qualification and training plans and other related Safeguards Information are protected against unauthorized disclosure in accordance with the requirements of Section 73.21 of this chapter, as appropriate. 82

EVALUATION AND CONCLUSION

ZPS-1 complies with this section.

EFFECTIVE DATE

Amended effective December 30, 1981 (46 F.R. 63031).

REFERENCES

1. FSAR Chapter 13.0, various subsections.

2. FSAR Position 432.2.

3. FSAR Section 6.2. 77

4. FSAR Positions 042.3, 041.40, and 041.47.

5. Draft ZPS-1 Emergency Response Plan.

(2) An evaluation of compliance is provided for each of the 10 CFR 50.55a paragraphs (c), (d), (e), (f), (g), and (i) referenced in paragraph (a)(2) above.

STATEMENT OF SECTION 50.55a(b)

The ASME Boiler and Pressure Vessel Code, which is referenced in the following paragraphs, was approved for incorporation by reference by the Director of the Federal Register on January 1, 1981. A notice of any changes made to the material incorporation by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H St. NW., Washington, D.C.

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions through the 1980 Edition and addenda through the Winter 1980 Addenda. |82 |82

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1 and include editions through the 1980 Edition and addenda through the Winter 1980 Addenda, subject to the following limitations and modifications: 77 |82

(i) "Limitations on specific editions and addenda." When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (6)(2) of this section, are not affected by their limitations. |82 |82

(ii) Pressure-retaining welds in ASME Code Class 1 piping (applies to Tables IWB-2500 and IWB-2500-1 and Category B-J). If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB-2500 and Table IWB-2600 Category B-J of Section XI of the ASME Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.

(iii) Steam generator tubing (modifies Article IWB-2000). If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000, the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications. |82

total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency. These classes are further discussed in NUREG-0654; FEMA-REP-1.

D. Notification Procedures

1. Administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary, shall be described. This description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZ's.

2. Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs. 77

3. A licensee shall have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that the State/local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition. By February 1, 1982, the nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The 4-month period in 10 CFR 50.54(s) (2) for the correction of emergency plan deficiencies shall not apply to the initial installation of this public notification system that is required by February 1, 1982. The 4-month period will apply to correction of deficiencies identified 82 82

during the initial installation and testing of the prompt public notification systems as well as those deficiencies discovered thereafter. The design objective of the prompt public notification shall be to have the capability to essentially complete the initial notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this notification capability will range from immediate notification of the public (within 15 minutes of the time that State and local officials are notified that a situation

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All training, including exercise, shall provide for formal critiques in order to identify weak areas that need corrections. Any weaknesses that are identified shall be corrected.

G. Maintaining Emergency Preparedness

Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.

H. Recovery

Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.

V. Implementing Procedures

No less than 180 days prior to scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material, 3 copies of each of the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Director of the appropriate NRC Regional Office with 10 copies to the Director of Nuclear Reactor Regulation or, if appropriate, the Director of Nuclear Material Safety and Safeguards. In cases where a decision on an operating license is scheduled less than one year after the effective date of this rule, such implementing procedures shall be submitted as soon as practicable but before full power operation is authorized. Prior to March 1, 1981, licensees who are authorized to operate a nuclear power facility shall submit 3 copies each of the licensee's emergency plan implementing procedures to the Director of the appropriate NRC Regional Office with 10 copies to the Director of Nuclear Reactor Regulation. Three copies each of any changes to maintain these implementing procedures up to date shall be submitted to the same NRC Regional Office with 10 copies to the Director of Nuclear Reactor Regulation or, if appropriate, the Director of Nuclear Material Safety and Safeguards within 30 days of such changes. 77

EFFECTIVE DATE

Amended effective December 30, 1981 (46 F.R. 63031). 82

EVALUATION OF COMPLIANCE

The ZPS-1 Emergency Plan was submitted to the NRC as Appendix F to the FSAR and was in compliance with the regulations in effect at the time of its submittal.

In response to the August 19, 1980 rule change to Appendix E, CG&E modified the Emergency Plan and submitted the new draft plan

to the NRC for review by letters from E. A. Borgmann (CG&E) to H. Denton (NRC) dated January 30, 1981 and July 1, 1981. The draft Emergency Plan is currently under NRC review and will be modified, as necessary, to bring it into compliance with this appendix. When the draft plan is approved, it will be submitted in FSAR Appendix F. 82

Emergency Plan implementing procedures will be submitted to the NRC as required by Part V. 77

CONCLUSION

ZPS-1 complies with this regulation.