Chapter 1: Introduction and Summary

Table of Contents

Chapter 1: Introduction and Summary

Table of Contents (continued)

Chapter 1: Introduction and Summary

Table of Contents (continued)

Chapter 1: Introduction and Summary

Table of Contents (continued)

Chapter 1: Introduction and Summary

List of Tables

Intentionally Blank

Chapter 1
INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This FSAR supports the operation of two similar nuclear power units, designated as Surry Power Station Units 1 and 2, constructed on a site situated on Gravel Neck and adjacent to the James River in Surry County, Virginia, pursuant to the construction permit issued by the Commission.

Each unit includes a pressurized water reactor (PWR) nuclear steam supply system and turbine generator furnished by Westinghouse Electric Corporation, similar in design concept to several projects licensed by the Commission. The balance of each unit was designed by Vepco, with the assistance of its agent, Stone & Webster Engineering Corporation.

Each reactor unit was designed for a warranted power output of 2441 MWt, with an equivalent warranted gross electrical output of 822.6 MWe. However, the nominal core power rating for each unit is 2546 MWt. All steam and power conversion equipment, including the turbine generator, has been designed on the basis of this higher thermal output and has the capability to generate a maximum calculated gross output of 855.4 MWe. The engineered safeguards systems and the containment are designed and evaluated for operation at this higher power level, which is used in the analysis of all postulated incidents in this report that have offsite consequences.

Unit 1 achieved commercial operation in December 1972 and Unit 2 in May 1973. In 1995, both units were uprated to a core power output of 2546 MWt (corresponding to a nuclear steam supply system power rating of 2558 MWt).

The remainder of Chapter 1 of this report summarizes the principal design features and safety criteria of the nuclear units by emphasizing the similarities and differences with respect to other pressurized water nuclear power plants at other sites.

Chapter 2 contains a description and evaluation of the Surry site and its environs and demonstrates the suitability of the site for reactors of the size and type described. Chapters 3 and 4 describe the reactor and the reactor coolant system, and Chapters 5 and 15 describe the containment structure and related systems. Chapters 7 through 11 describe the other auxiliary systems. Chapters 5, 6, 7, 8, and 9 include descriptions of the various systems directly related to safeguards. Chapter 12 reviews Vepco's organization and technical competence, associated contractors and consultants, and information relating to station organization and personnel training. Chapter 13 describes Vepco's approach to initial tests and operation. Chapter 14 relates to safety evaluation; it summarizes the analyses that demonstrate the adequacy of the reactor protection system, the containment system, and the engineered safeguards system, and shows that the consequences of various postulated incidents are within the guidelines suggested in the

Commission's regulation 10 CFR 100, 10 CFR 50.67, or Regulatory Guide 1.183 (RG 1.183). Chapter 17 describes the quality assurance program for the operational phase of Vepco's nuclear power stations. Chapter 18 describes the aging management programs and activities credited in support of the renewed operating license 20-year period of extended operation. The inclusion of Chapter 18 into the UFSAR is a condition of the renewed operating licenses. This final safety analysis report has been prepared using the AEC publication *A Guide for the Organization and Contents of Safety Analysis Reports* as a guide. Refinements may be made from time to time through amendments to this report.

With respect to the numbers, graphs, and drawings included within this report, it should be understood that normal tolerance permitted by good engineering practice is intended. Where operating parameters are unusually important, it is acknowledged that such items are included in the Technical Specifications, the adoption of which is a condition of the operating license.

1.1.1 Design Highlights

The design of the Surry Power Station is based upon concepts that have been developed and successfully applied in the construction of other PWR systems. In subsequent paragraphs, certain design features of the Surry Power Station are indicated that represent slight variations or extrapolations from other units approved for operation, such as H. B. Robinson 2 (Docket 50-261) and Indian Point 3 (Docket 50-286).

1.1.2 Power Level

The nominal power rating for each unit of the Surry Power Station is set at 2546 MWt. Site and engineered safeguards evaluation has been performed for a reactor thermal output of 2546 MWt, which corresponds to the maximum calculated nominal rating of the turbine generator. The 2546-MWt power rating is achieved by about a 4.3% increase in the average reactor heat flux over the 2441-MWt rating established for initial operation.

1.1.3 Reactor Coolant Loops

The reactor coolant system for each unit consists of three loops, each loop having components (steam generator, pumps, and piping) similar to those at Indian Point Unit 2, except that each of the Surry units has two reactor coolant loop stop valves and a bypass valve in each loop.

1.1.4 Peak Specific Power

The operation of the initial core cycle at 2441 MWt yielded a maximum steady-state peak specific power of 17.3 kW/ft and a corresponding peak power of 19.4 kW/ft for the 112% overpower condition. These values were justified by the results of incore experiments by Westinghouse and others at these and higher specific power ratings. These ratings were lower than the corresponding conditions for Indian Point Unit 2, which were 18.4 kW/ft steady-state and 20.6 kW/ft overpower, and which were a result of lower design hot-channel factors.

1.1.5 Fuel Clad

The fuel rod design for each unit uses Zircaloy-4 or ZIRLO as a clad material. Zircaloy-4 has proved successful in the Carolinas-Virginia Tube Reactor (CVTR) and Saxton reactors, in Yankee (Rowe) test assemblies, and it is used in many Westinghouse reactors. ZIRLO has also been successfully used in several reactors, including V.C. Summer, Indian Point 2 and North Anna Units 1 and 2.

1.1.6 Fuel Assembly Design

The fuel assembly incorporated the rod cluster control concept in a canless 15 x 15 fuel and control rod array using grids to provide support for the fuel rods. Extensive out-of-pile tests have been performed on this concept, successful in-pile tests have been performed in the Saxton reactor, and operating experience is available from the San Onofre, Connecticut Yankee, and other similar plants. Prior to the introduction of Surry Improved Fuel (SIF) for both units, all grids were made of Inconel. Beginning with SIF, all intermediate spacer grids will be made of either Zircaloy or ZIRLO.

1.1.7 Moderator Temperature Coefficient of Reactivity

Burnable poison rods are used in the reactor unit to provide a negative moderator temperature coefficient at cycle start-up. As the fuel in the core is depleted and the boron shim concentration is decreased, the moderator temperature coefficient becomes more negative.

1.1.8 Containment

The reactor containment concept is based on the use of a reinforced-concrete container structure similar to that of the Connecticut Yankee Atomic Power Plant, but the containment is maintained at subatmospheric pressure during normal operation. Following the postulated loss-of-coolant accident (LOCA) described in Chapter 14, the containment peak pressure would be reduced to subatmospheric by the use of redundant chemical spray cooling systems, thereby positively terminating outleakage to the environment within 1 hour after the initiation of the accident assuming the most limiting single failure, i.e., loss of emergency power to one train of spray systems. These original design criteria were modified in conjunction with the analyses for implementation of the alternative source term. The modified criteria require that, following the LOCA, the containment pressure be less than 0.5 psig for Unit 1 (1.0 psig for Unit 2) within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for Unit 1 (1.0 psig for Unit 2) for the interval from 1 to 4 hours following the Design Basis Accident. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

1.1.9 Xenon Oscillations

Ex-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon-induced power oscillations. Extensive analysis, with confirmation of methods by special transient experiments at Haddam Neck, has shown that any induced radial or diametral xenon transients would die away naturally. A full discussion of xenon stability control can be found in WCAP 7208-L (1968), *Power Distribution Control of Westinghouse Pressurized Water Reactors*, Westinghouse proprietary.

1.2 SUMMARY

1.2.1 General

Each unit at the Surry Power Station incorporates a closed-cycle pressurized water nuclear steam supply system, a turbine generator, and their necessary auxiliaries. Radioactive waste disposal systems, a fuel handling system, and all auxiliaries, structures, and other onsite facilities required for a complete and operable nuclear power station are also provided. The general arrangement of the units is shown in the site plan, Figure 15.1-1, and the plot plan, Reference Drawing 1.

1.2.2 Structures

The major structures are the reactor containments, auxiliary building, fuel building, turbine building, and service building, which includes the main control area. General layouts of the reactor containment for Unit 1, the auxiliary building, and the fuel building, showing interior arrangements, are given on Reference Drawings 2 through 14.

Each reactor containment is a steel-lined, reinforced-concrete cylinder with a hemispherical dome and a flat, reinforced-concrete foundation mat. Each containment is designed to withstand the internal pressure accompanying the hypothetical design-basis incident, is virtually leaktight, and provides adequate radiation shielding for both normal operation and design-basis accident (DBA) conditions. Whenever at subatmospheric pressure, there is no outleakage of activity from the containment structure. The seismic criteria used in the design of the structures and equipment in the station are described in Section 2.5. The maximum horizontal ground acceleration for design purposes is 0.07g. The design-basis maximum horizontal ground acceleration is assumed to be 0.15g. Dampening at these accelerations has been assumed to be 5% and 10%, respectively. Vertical acceleration is two-thirds of the horizontal acceleration and is considered to act simultaneously with the horizontal acceleration.

1.2.3 Nuclear Steam Supply System

The nuclear steam supply system for each unit consists of a pressurized water reactor, a reactor coolant system, and associated auxiliary systems. The reactor coolant system is arranged as three closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump, isolation and bypass valves, piping, and a steam generator. An electrically heated pressurizer is connected to one of the loops.

Each reactor core includes uranium dioxide pellets, enclosed in zirconium alloy tubes with welded end plugs, as fuel. The tubes are supported in assemblies by structures of grids and there are suitable end pieces for the support of the assembled rods and restraint of abnormal axial movement. The mechanical control rod assemblies consist of clusters of stainless-steel-clad absorber rods that are guided by tubes located within the fuel assembly. The core consists of 157 of these fuel assemblies loaded in varying enrichments. Originally, an out-in fuel management approach was used in core design. Fresh, high-enrichment fuel was introduced into the core outer region. At the next refueling, it was moved to the core inner region where it was intermingled with fuel moved from the outer region during the previous refueling. Two refuelings later, the original high-enrichment fuel was discharged to spent-fuel storage. Currently a low leakage type of fuel management is employed which places burned fuel assemblies on the core periphery and intermingles the fresh fuel assemblies with previously burned assemblies in the core's interior regions.

The steam generators are vertical U-tube units containing Inconel tubes. Integral separating equipment reduces the moisture content of the steam at the turbine throttle to 0.25% or less.

The reactor coolant pumps are vertical, single-stage, centrifugal pumps equipped with controlled-leakage shaft seals.

The reactor coolant loop stop and bypass valves are motor-operated gate valves that are remotely controlled from the control room. These valves permit any loop to be isolated from the reactor vessel.

Nuclear auxiliary systems are provided to perform the following functions:

1. Accommodate reactor coolant system water makeup requirements.
2. Purify reactor coolant water.
3. Introduce chemicals for corrosion inhibition.
4. Introduce and remove chemicals for reactivity control.
5. Cool system components.
6. Remove residual heat during a portion of the reactor cooling period and also when the reactor is shut down.
7. Cool the spent-fuel pool water.
8. Permit the sampling of reactor coolant water.
9. Provide for emergency safety injection.
10. Vent and drain the reactor coolant system and the auxiliary systems.
11. Provide emergency containment spray.
12. Provide emergency chemical containment spray.
13. Maintain a subatmospheric containment pressure.
14. Provide containment ventilation and cooling.
15. Dispose of liquid and gaseous wastes, and provide for the disposal of solid wastes.

1.2.4 Reactor and Station Controls

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rod assemblies. The control system permits the unit to accept step load increases of 10% and ramp load increases of 5% per minute over a load range of 15% to 100% power under normal operating conditions, subject to xenon limitations.

The control of both the reactor and turbine generator for each unit is accomplished from the control room and is supervised by licensed operators.

1.2.5 Waste Disposal System

The waste disposal system provides all equipment necessary to collect, process, and prepare for disposal of all radioactive liquid, gaseous, and solid waste produced as a result of station operation. The waste disposal system is capable of handling the wastes produced by both units as a result of station operation.

Liquid wastes are collected and processed by evaporation, reverse osmosis, and/or ion exchange. Processed liquid is analyzed before discharge into the river. Discharges are maintained below limits established by 10 CFR 20 or other appropriate regulations. Non-combustible and combustible solid wastes can be sorted, shredded, baled or drummed consistent with applicable offsite processing or disposal requirements. They are shipped from the site for ultimate disposal at an authorized location.

Gaseous wastes are diluted and discharged to the environment with a yearly average radioactivity level within the limits set forth in 10 CFR 20.

1.2.6 Fuel Handling Systems

The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves a reactor vessel until it is placed in a cask for shipment from the site. Spent fuel is transferred under water, which provides an optically transparent radiation shield and a reliable source of coolant for removal of residual heat.

1.2.7 Turbines and Auxiliaries

Each turbine is a tandem-compound, three-element, 1800-rpm unit having 44-inch last-stage exhaust blading in the low-pressure elements. Four combination moisture separators-reheaters are employed to dry and superheat the steam between the high-pressure and low-pressure turbine cylinders for each unit. A single-pass, deaerating surface condenser installed in two sections, two 100%-capacity steam jet air ejectors, three 50%-capacity condensate pumps, two 50%-capacity steam generator feedwater pumps, three auxiliary feedwater pumps, and six stages of feedwater heating are provided.

1.2.8 Electrical Systems

The main generator for each unit is an 1800-rpm, 22-kV, three-phase, 60-cycle, hydrogen inner-cooled unit. A main step-up transformer delivers power to the high-voltage switchyard.

The station service power distribution system for each unit consists of station service transformers, 4160V and 480V switchgear and buses, and 480V motor control centers. The normal source of station service power is the main generator, while the reserve station service transformers provide an alternate source via the switchyard. The emergency power distribution system consists of 4160V and 480V switchgear and buses, 480V motor control centers, 120V ac vital buses, and 125V dc batteries and equipment. The emergency buses are normally powered from the switchyard via the three reserve station service transformers.

Emergency power is supplied by alternate sources, including one emergency diesel-driven generator for each unit and a third diesel-driven generator shared by both units. Each diesel-driven generator is capable of operating post-incident containment recirculation spray pumps as well as charging pumps and low-head safety injection pumps to ensure an acceptable containment pressure transient during the design-basis accident.

1.2 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11448-FY-1D	Plot Plan
2.	11448-FM-1A	Machine Location: Reactor Containment, Elevation 47'- 4"
3.	11448-FM-1B	Machine Location: Reactor Containment, Elevation 18'- 4"
4.	11448-FM-1C	Machine Location: Reactor Containment, Elevation 3'- 6"
5.	11448-FM-1D	Machine Location: Reactor Containment, Elevation 27'- 7"
6.	11448-FM-1E	Machine Location: Reactor Containment; Sections “A-A”, “E-E”, & “Z-Z”
7.	11448-FM-1F	Machine Location: Reactor Containment; Sections “B-B”, “X-X”, & “Y-Y”
8.	11448-FM-1G	Machine Location: Reactor Containment, Sections “C-C” & “D-D”
9.	11448-FM-5A	Arrangement: Auxiliary Building
10.	11448-FM-5B	Arrangement: Auxiliary Building, Unit 1
11.	11448-FM-5C	Arrangement: Auxiliary Building
12.	11448-FM-5D	Arrangement: Auxiliary Building
13.	11448-FM-9A	Arrangement: Fuel Building, Sheet 1
14.	11448-FM-9B	Arrangement: Fuel Building, Sheet 2, Unit 1

Intentionally Blank

1.3 COMPARISON WITH OTHER STATIONS

Table 1.3-1 presents a summary of the design and operating parameters for the Surry Power Station nuclear steam supply systems. The table provides a comparison of these data with the data available from the FSAR of Turkey Point Units 3 and 4, and with the data available from the FSAR of H. B. Robinson Unit 2.

The Turkey Point and H. B. Robinson references were selected because both are closely related technically to the Surry units, and both were reviewed for operating licenses during the same general time frame as the Surry units.

The Surry Power Station units are also generally comparable with the pressurized water reactors at Pacific Gas and Electric Company, Duquesne Light Company, Rochester Gas and Electric Company, and the Tennessee Valley Authority (Sequoyah Units).

Table 1.3-1
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Thermal and Hydraulic Design Parameters			
Total core heat output	2441 MWt	2200 MWt	2200 MWt
Total core heat output	8331×10^6 Btu/hr	7479×10^6 Btu/hr	7479×10^6 Btu/hr
Heat generated in fuel	97.4%	97.4%	97.4%
Maximum thermal overpower	112%	112%	112%
System operating pressure, nominal	2250 psia	2250 psia	2250 psia
System operating pressure, minimum steady state	2200 psia	2200 psia	2200 psia
DNB ratio at nominal initial rating condition	1.97	1.81	1.81
Minimum DNB ratio for design transients	1.30	1.30	1.30
Coolant flow			
Total flow rate	100.7×10^6 lb/hr	101.5×10^6 lb/hr	101.5×10^6 lb/hr
Effective flow rate for heat transfer	96.2×10^6 lb/hr	97.0×10^6 lb/hr	97.0×10^6 lb/hr
Effective flow area for heat transfer	41.8 ft^2	41.8 ft^2	41.8 ft^2
Average velocity along fuel rods	14.2 ft/sec	14.3 ft/sec	14.3 ft/sec
Average mass velocity	2.31×10^6 lb/hr-ft^2	2.32×10^6 lb/hr-ft^2	2.32×10^6 lb/hr-ft^2
Coolant temperature (at 100% power)			
Nominal inlet	543°F	546.2°F	546.2°F
Minimum inlet due to instrumentation error and deadband	547°F	550.2°F	550.2°F
Average rise in vessel	62.6°F	55.9°F	55.9°F
Average rise in core	65.5°F	58.3°F	58.3°F
Average in core	577°F	575.4°F	575.4°F
Average in vessel	574°F	574.2°F	574.2°F
Nominal outlet of hot channel	642°F	642°F	642°F

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Thermal and Hydraulic Design Parameters (continued)			
Average film coefficient	5400 Btu/hr-ft^2-°F	5400 Btu/hr-ft^2-°F	5400 Btu/hr-ft^2-°F
Average film temperature difference	35.0°F	31.8°F	31.8°F
Heat transfer at 100% power			
Active heat transfer surface area	42,460 ft^2	42,460 ft^2	42,460 ft^2
Average heat flux	191,100 Btu/hr-ft^2	171,600 Btu/hr-ft^2	171,600 Btu/hr-ft^2
Maximum heat flux	534,100 Btu/hr-ft^2	554,200 Btu/hr-ft^2	554,200 Btu/hr-ft^2
Average thermal out put	6.2 kW/ft	5.5 kW/ft	5.5 kW/ft
Maximum thermal out put	17.3 kW/ft	17.9 kW/ft	17.9 kW/ft
Maximum clad surface temperature at nominal pressure	657°F	657°F	657°F
Fuel central temperature			
Maximum at 100% power	4,050°F	4,030°F	4,030°F
Maximum at overpower	4,300°F	4,300°F	4,300°F
Thermal output, at maximum overpower	20.6 kW/ft	20.0 kW/ft	20.0 kW/ft
Core Mechanical Design Parameters			
Fuel assemblies			
Design	Canless 15 x 15	Canless 15 x 15	Canless 15 x 15
Rod pitch	0.563 in.	0.563 in.	0.563 in.
Overall dimensions	8.426 x 8.426 in.	8.426 x 8.426 in.	8.426 x 8.426 in.
Fuel weight (as UO_2)	176,200 lb	176,200 lb	176,200 lb
Total weight	226,200 lb	226,200 lb	226,200 lb
Number of grids per assembly	7	7	7

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Core Mechanical Design Parameters (continued)			
Fuel rods			
Number	32,028	32,028	32,028
Outside diameter	0.422 in.	0.422 in.	0.422 in.
Diametral gap	0.0075/0.0075/0.0085 in.	0.0065 in.	0.0065 in.
Clad thickness	0.0243 in.	0.0243 in.	0.0243 in.
Clad material	Zircaloy-4	Zircaloy	Zircaloy
Fuel pellets			
Material	UO^2 sintered	UO^2 sintered	UO^2 sintered
Density (percent of theoretical)	94, 93, 92	94, 92, 91	94, 92, 91
Diameter	0.3659/0.3659/0.3649 in.	0.3669 in.	0.3669 in.
Length	0.6000 in.	0.6000 in.	0.6000 in.
Control rod assemblies			
Neutron adsorber	5% Cd, 15% In, 80% Ag	5% Cd, 15% In, 80% Ag	5% Cd, 15% In, 80% Ag
Cladding material	SS 304, cold worked	SS 304, cold worked	SS 304, cold worked
Clad thickness	0.019 in.	0.019 in.	0.019 in.
Number of control rod assemblies (full/part length)	48/5	45/8	45/8
Number of rods per assembly	20	20	20
Total rod worth (k per k)	See Table 3.3-3	See Chapter 3 of Turkey Point FSAR	See Chapter 3 of H.B. Robinson FSAR
Core structure			
Core barrel i.d./o.d.	133.875/137.875 in.	133.875/137.875 in.	133.875/137.875 in.
Thermal shield i.d./o.d.	142.625/148.000 in	142.625/148.0 in	142.625/148.0 in

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
	Nuclear Design Data		
Structural characteristics			
Fuel weight (as UO_2)	175,600 lb	176,200 lb	176,200 lb
Clad weight	36,300 lb	36,300 lb	36,300 lb
Core diameter (equivalent)	119.5 in.	119.5 in.	119.5 in.
Core height (active fuel)	144 in.	144 in.	144 in.
Reflector thickness and composition			
Top - water plus steel	10 in.	10 in.	10 in.
Bottom - water plus steel	10 in.	10 in.	10 in.
Side - water plus steel	15 in.	15 in.	15 in.
H_2O/U, unit cell (cold volume ratio)	4.18	4.18	4.18
Number of fuel assemblies	157	157	157
Uranium dioxide rods per assembly	204	204	204
Performance characteristics			
Loading technique	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform
Fuel burnup, MWD/MTU			
Average first cycle	12,600	13,000	13,000
First core average	22,300	24,500	24,500
Feed enrichments, wt.%			
Region 1	1.85	1.85	1.85
Region 2	2.55	2.55	2.55
Region 3	3.10	3.10	3.10

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Nuclear Design Data (continued)			
Control characteristics			
Effective multiplication (beginning of life), k_{eff}			
Cold, no power, clean	1.176	1.180	1.180
Hot, no power, clean	1.145	1.38	1.38
Hot, full power, Xe and Sm equilibrium	1.090	1.077	1.077
Boron concentrations			
To shut reactor down with no control rod assemblies inserted, clean (k_{eff} = 0.99)			
Cold	1250 ppm	1250 ppm	1250 ppm
Hot	1240 ppm	1210 ppm	1210 ppm
To control at power with no control rod assemblies inserted, clean/equilibrium Xe and Sm	1005 ppm/705 ppm	1000 ppm/670 ppm	1000 ppm/920 ppm
Burnable poison worth, hot	6.9% k/k	7.3% k/k	7.3% k/k
Burnable poison worth, cold	5.3% k/k	5.6% k/k	5.6% k/k
Kinetic characteristics			
Moderator temperature coefficient	$+0.3 \times 10^{-4}$ to -3.5×10^{-4} k/k per °F	$+0.3 \times 10^{-4}$ to -3.5×10^{-4} k/k per °F	10.3×10^{-4} to -3.5×10^{-4} k/k per °F
Moderator pressure coefficient	-0.3×10^{-6} to $+3.5 \times 10^{-6}$ k/k per psi	-0.3×10^{-6} to $+3.4 \times 10^{-6}$ k/k per psi	-0.3×10^{-6} to $+3.5 \times 10^{-6}$ k/k per psi
Moderator density (void) coefficient	–0.1 to +0.3 k/k per g per cm^3	$+0.5 \times 10^{-3}$ to -2.5×10^{-3} k/k per °F	$+0.5 \times 10^{-3}$ to -2.5×10^{-3} k/k per % void
Doppler coefficient	-1×10^{-5} to -1.6×10^{-5} k/k per °F	-1×10^{-5} to -1.6×10^{-5} k/k per °F	-1×10^{-5} to -1.6×10^{-5} k/k per °F

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Reactor Coolant System—Code Requirements			
Component			
Reactor vessel	ASME III, Class A	ASME III, Class A	ASME III, Class A
Steam generator			
Tube side	ASME III, Class A	ASME III, Class A	ASME III, Class A
Shell side	ASME III, Class C	ASME III, Class C	ASME III, Class C
Pressurizer	ASME III, Class A	ASME III, Class A	ASME III, Class A
Pressurizer relief tank	ASME III, Class C	ASME III, Class C	ASME III, Class C
Pressurizer safety valves	ASME III	ASME III	ASME III
Reactor coolant piping	USAS B31.1	USAS B31.1	USAS B31.1
Principal Design Parameters of the Reactor Coolant System (100%)			
Reactor core heat output	2441 MWt	2200 MWt	2200 MWt
Reactor heat output	8331×10^6 Btu/hr	7479×10^6 Btu/hr	7479×10^6 Btu/hr
Operating pressure	2235 psig	2235 psig	2235 psig
Reactor inlet temperature	543°F	546.2°F	546.2°F
Reactor outlet temperature	605.8°F	602.1°F	602.1°F
Number of loops	3	3	3
Design pressure	2485 psig	2485 psig	2485 psig
Design temperature	650°F	650°F	650°F
Hydrostatic test pressure (cold)	3107 psig	3107 psig	3107 psig
Coolant volume, including total pressurizer	9458 ft^3	9088 ft^3	9088 ft^3
Total reactor flow	265,500 gpm	268,500 gpm	2,68,500 gpm

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Reactor Design Parameters of the Reactor Vessel			
Material	SA-302 Grade B, low-alloy steel, internally clad with austenitic SS 304	SA-302 Grade B, low-alloy steel, internally clad with austenitic SS 304	SA-302 Grade B, low-alloy steel, internally clad with austenitic SS 304
Design pressure	2485 psig	2485 psig	2485 psig
Design temperature	650°F	650°F	650°F
Operating pressure	2235 psig	2235 psig	2235 psig
Inside diameter of shell	157 in.	155.5 in.	155.5 in.
Outside diameter across nozzles	252 in.	236 in.	236 in.
Overall height of vessel and enclosure head	40 ft. 5 in.	41 ft. 6 in.	41 ft. 6 in.
Minimum clad thickness	5/32 in.	5/32 in.	5/32 in.
Principal Design Parameters of the Steam Generators			
Number of units	3	3	3
Type	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator
Tube material	Inconel	Inconel	Inconel
Shell material	Carbon steel	Carbon steel	Carbon steel
Tube side design pressure	2485 psig	2485 psig	2485 psig
Tube side design temperature	650°F	650°F	650°F
Tube side design flow	33.57×10^6 lb/hr	33.43×10^6 lb/hr	33.93×10^6 lb/hr
Shell side design pressure	1085 psig	1085 psig	1085 psig
Shell side design temperature	600°F	556°F	556°F
Operating pressure, tube side, nominal	2235 psig	2235 psig	2235 psig
Operating pressure, shell side, maximum	1005 psig	1005 psig	1005 psig

Table 1.3-1 (continued)
COMPARISON OF INITIAL DESIGN PARAMETERS

Design Parameter	Surry Power Station	Turkey Point 3 or 4	H.B. Robinson
Principal Design Parameters of the Steam Generators (continued)			
Maximum moisture at outlet at full load	0.25%	0.25%	0.25%
Hydrostatic test pressure, tube side (cold)	3107 psig	3107 psig	3110 psig
Principal Design Parameters of the Reactor Coolant Pumps			
Number of units	3	3	3
Type	Vertical single-stage mixed flow with bottom suction and horizontal discharge	Vertical single-stage radial flow with bottom suction and horizontal discharge	Vertical single-stage radial flow with bottom suction and horizontal discharge
Design pressure	2485 psig	2485 psig	2485 psig
Design temperature	650°F	650°F	650°F
Operating pressure, nominal	2235 psig	2235 psig	2235 psig
Suction temperature	543°F	546.5°F	546.5°F
Design capacity	88,500 gpm	89,500 gpm	89,500 gpm
Design head	280 ft	260 ft	260 ft
Hydrostatic test pressure (cold)	3107 psig	3107 psig	3107 psig
Motor type	Single-speed ac induction	Single-speed ac induction	Single-speed ac induction
Motor rating	6000 hp	6000 hp	6000 hp
Principal Design Parameters of the Reactor Coolant Piping			
Material	Austenitic SS	Austenitic SS	Austenitic SS
Hot leg, i.d.	29 in.	29 in.	29 in.
Cold leg, i.d.	27.5 in.	27.5 in.	27.5 in.
Between pump and steam generator, i.d.	31 in.	31 in.	31 in.
Design pressure	2485 psig	2485 psig	2485 psig

Intentionally Blank

1.4 COMPLIANCE WITH CRITERIA

The design of the Surry Power Station meets the intent of the criteria as expressed within this section. Following the text of each criterion is a brief discussion specific to that criterion.

1.4.1 Quality Standards

> Those systems and components of reactor facilities that are essential to the prevention of accidents that could affect the public health and safety or to the mitigation of their consequences are designed, fabricated, and erected in accordance with quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they are identified. Where adherence to such codes or standards does not suffice to ensure a quality product in keeping with the safety function, they are supplemented or modified as necessary. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Refer to the response to the criterion in Section 1.4.2.

1.4.2 Performance Standards

> Those systems and components of reactor facilities that are essential to the prevention of accidents that could affect the public health and safety or to the mitigation of their consequences are designed, fabricated, and erected in accordance with performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established reflect (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area, and (b) an appropriate margin for withstanding forces greater than those recorded, in view of uncertainties about the historical data and their suitability as a basis for design.

Those features of reactor facilities essential to the prevention of accidents that could affect the public health and safety or to the mitigation of their consequences are designed, fabricated, and erected in conformity with:

1. Quality standards that reflect the importance of the safety function to be performed. Approved design codes are used when appropriate to the nuclear application.
2. Performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquake, flooding condition, wind, ice, or other natural phenomena characteristic of the site.

Features of the facility essential to accident prevention and the mitigation of accident consequences are the designs of the fuel, reactor coolant, and containment barriers; the controls and emergency cooling systems whose function is to maintain the integrity of these three barriers;

systems that depressurize the containment following a LOCA; a power supply and essential services; and the components employed to safely convey and store radioactive wastes and spent reactor fuel.

The fuel assembly rod design considers the effect on the zirconium alloy cladding of internal fission gas pressure buildup, thermal expansion, irradiation, and fabrication variations. Core design conditions and cladding material specifications are selected to limit hydrogen absorption during core life to levels that do not affect fuel cladding integrity. To ensure high quality, fuel rod materials are subjected to chemical analysis and tensile tests and the rods receive dimensional inspection, X-ray of welds, ultrasonic tests, and helium leak tests.

Quality standards of material selection, design, fabrication, and inspection governing the above features conform to the applicable provisions of recognized codes and good nuclear practice. The reinforced-concrete reactor containment structures conform to the applicable portions of ACI-318-63.

Further elaboration on quality standards of the reactor containment is given in Chapter 15. Vessels comply with Section III of the ASME Code under the specific classification dictated by their use. The principles of this Code, or equivalent guidelines, are employed where the Code is not strictly applicable but where the safety function calls for an equivalent assurance of quality. In the same manner, piping conforms to the requirements of USAS B31.1.

Particular emphasis is placed on the assurance of quality of each reactor vessel and hence on the acquisition of materials whose properties are uniformly within tolerances appropriate to the application of the design methods of the Code. The fatigue usage factor, derived from an assumed number of thermal cycles that is probably more than four times the number of such cycles actually expected, is less than that at which the propagation of material defects would occur.

The design margin and material surveillance ensure that each vessel is operated well within the ductile range of temperatures when the reactor vessel is operated within established operational limits. Further discussion of quality assurance for the reactor vessels, including the use of vessel irradiation test specimens, is given in Chapter 4.

All piping, components, and supporting structures of each reactor and the safety-related systems are designed to withstand a specified seismic disturbance in excess of that predicted for the site. Station design criteria specify that there is no loss of function of such equipment in the event of the DBA ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of Class I structures to ground acceleration, based on appropriate characteristics of the site foundation soils and on the critical damping of the foundations and structures, is included in the design analysis.

Each reactor containment is defined for seismic purposes as a Class I structure. Structural members have sufficient capacity to accept, without exceeding yield stresses, a combination of normal operating loads, functional loads due to a design-basis accident and the loadings imposed by the maximum wind velocity or the design-basis earthquake (DBE), whichever is larger. The emergency onsite power sources are not subject to interruption due to earthquakes, windstorms, floods, or disturbances in the external power transmission system. Power cabling, motors, and other equipment required for the operation of the engineered safeguards is suitably protected against the effects of the design-basis accident and other severe external environmental conditions, as applicable, to ensure a high degree of confidence in the operability of these systems should they be required.

The reference chapters are as follows:

Title	Chapter
Site	2
Reactor	3
Reactor Coolant System	4
Containment System	5
Engineered Safeguard	6
Instrumentation and Control	7
Electrical Systems	8
Auxiliary and Emergency Systems	9
Radioactive Wastes and Radiation Protection	11
Structures and Construction	15

1.4.3 Fire Protection

The reactor facility is designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events on safety. Non-combustible and fire-resistant materials are used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as the containment, the control room, and components of engineered safeguards.

Fire or explosions occurring within the reactor facility are avoided because of the inherent preventive features in the station design.

Waste hydrogen gas is collected in the waste gas decay tanks. The oxygen content of the tank is limited administratively to 2% by volume. The oxygen content by volume may be diluted to a concentration below its upper limit by the addition of nitrogen to the tank (preferred - to maximize radioactivity decay time of waste gases) or by performing a release. Systems processing hydrogen-oxygen mixtures that are potentially hazardous conform to the National Electrical Code for Areas of Class I, Division 2, Group B. All spark-producing devices near the waste hydrogen equipment are explosion-proof.

The containment and other structures containing safe-shutdown equipment are of fire resistive or non-combustible construction and contain mostly non-combustible equipment. Atmospheric conditions inside the containment are not of an explosive nature.

The control room is of non-combustible construction and is isolated from surrounding areas by heavy concrete shielding. The control room atmosphere is not explosive and is maintained under positive pressure by its air conditioning system.

The references chapters are as follows:

Title	Chapter
Reactor Coolant System	4
Containment System	5
Auxiliary and Emergency System	9
Radioactive Wastes and Radiation Protection	11

1.4.4 Sharing of Systems

> Reactor facilities do not share systems or components unless it is shown that safety is not impaired by the sharing.

The facilities that have shared systems or components are tabulated in Section 1.5, with references to sections containing specific design details.

No impairment of the safety of the reactor facilities is caused by the sharing of any of these systems, and in certain instances such sharing enhances system reliability.

1.4.5 Records Requirements

> Records of the design, fabrication, and construction of essential components of the plant are maintained by the reactor operator or are under Vepco's control throughout the life of the reactor.

Records of the design, fabrication, and construction of essential components are maintained during the life of the unit and are available to Vepco. Chapter 17 includes a discussion of this matter. Records of all tests performed and test procedures used are kept by Vepco for the life of the unit.

The reference chapter is as follows:

Title	Chapter
Quality Assurance (Topical Report)	17

1.4.6 Reactor Core Design

> The reactor core with its related controls and protection systems is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits that have been stipulated and justified. The core and related auxiliary system designs provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations that can be anticipated.

The reactor core with its related control and protection system is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater, and loss of all offsite power.

The reactor control and protection instrumentation is designed to actuate a reactor trip for any anticipated combination of unit conditions when necessary to ensure a minimum departure from nucleate boiling ratio (DNBR) equal to or greater than the design DNBR limit (Section 3.2.3) and fuel center temperatures below the melting point of uranium dioxide.

The references chapters are as follows:

Title	Chapter
Reactor	3
Instrumentation and Control	7
Safety Analysis	14

1.4.7 Suppression of Power Oscillations

> The design of the reactor core with its related controls and protection systems ensures that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

The design of the reactor core and related protection systems ensures that power oscillations that could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core are reviewed as part of the core stability evaluation described in Section 1.6. Ex-core instrumentation is provided to obtain necessary information concerning axial and azimuthal power distributions. This instrumentation is adequate to enable the operator to monitor and control xenon-induced oscillations. Based on the deviations detected by the long ion chambers, provisions in the reactor control and protection system reduce trip setpoints and if necessary initiate load runback to maintain margin to departure from nucleate boiling as a result of these potential oscillations in power distribution. Incore instrumentation is used to periodically calibrate and verify the information provided by the ex-core instrumentation.

The general conclusion based on experimental results from SENA and San Onofre is that the ex-core instruments do give an accurate indication of the fact that power redistribution is taking place. This has been confirmed by a comparison with incore instrumentation results.

The temperature coefficient in the power operating range was maintained zero or negative by the inclusion of burnable poison shims in the initial core loading. Burnable poison shims can also be used in subsequent core loadings if necessary.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Instrumentation and Control	7

1.4.8 Overall Power Coefficient

> The reactor is designed so that the overall power coefficient in the power operating range is not positive.

The overall power coefficient is negative under normal operating conditions throughout core life.

The reference chapter is as follows:

Title	Chapter
Reactor	3

1.4.9 Reactor Coolant Pressure Boundary

> The reactor coolant pressure boundary is designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

The reactor coolant system, in conjunction with its control and protective provisions, is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation or anticipated system interactions, and to remain within the applicable code stress limits.

The fabrication of the components that constitute the pressure-retaining boundary of the reactor coolant system is carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for reactor coolant system components are more restrictive than applicable codes.

The materials of construction of the pressure-retaining boundary of the reactor coolant system are protected by the control of coolant chemistry so as to prevent corrosion phenomena that might otherwise reduce the system structural integrity during its service lifetime.

The reference chapter is as follows:

Title	Chapter
Reactor Coolant System	4

1.4.10 Containment

> Containment is provided. The containment structure is designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity, and, together with other engineered safeguards as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

A reinforced-concrete, steel-lined containment structure operating at subatmospheric pressure encloses the entire reactor coolant system. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the reactor coolant system. Engineered safeguards, which consist of safety injection systems and containment depressurization systems, serve to cool the reactor core and return the containment to subatmospheric pressure and maintain it at subatmospheric pressure for as long as the situation requires. The containment and its associated engineered safeguards exceed the required functional capability of protecting the public from the consequences of gross equipment failures, since they provide for a rapid termination of the effects of the event.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6

1.4.11 Control Room

> The facility is provided with a control room from which actions to maintain the safe operation of the plant can be controlled.
>
> Radiation protection is provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It is possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost through fire or other causes.

The control room is located at grade level in the service building. All safety-related switchgear, motor-generator sets, auxiliary instrument areas, battery rooms, and communications equipment are located in the basement of the service building. Sufficient shielding, distance, and containment integrity are provided to ensure that under postulated accident conditions during occupancy of the control room, control room personnel shall not be subjected to doses that, in the aggregate, would exceed the limits in 10 CFR 50.67. Emergency air-conditioning equipment is provided within the envelope of the shielded control room and associated portions of the basement, collectively called the control and relay room area. The control room is provided with the switchyard control panel, electrical recording panels, dc distribution panels, and a control panel for the operation of the diesel-generator system. The control panels contain those instruments and controls necessary for the operation of station and unit systems such as the reactor and its auxiliary systems, the turbine generator, and the steam and power conversion systems. Loading from the various station electrical distribution boards, such as the start-up boards, shutdown boards, and motor control centers, is accomplished from the station control panels.

The control room is common to the two units and is continuously occupied by qualified operating personnel under all operating and accident conditions.

In the event that access to the control room is restricted, either local control stations or the manual operation of critical components within the main control area can be used to effect hot shutdown from outside the control room.

The reference chapters are as follows:

Title	Chapter
Instrumentation and Control	7
Auxiliary and Emergency Systems	9

1.4.12 Instrumentation and Control Systems

> Instrumentation and controls are provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, primary coolant pressure and temperature, and control rod assembly positions within prescribed operating ranges.

The non-nuclear-regulating process and containment instrumentation measures temperatures, pressure, flow, and levels in the reactor coolant system, main steam system, containment, and auxiliary systems. Process variables required on a continuous basis for the start-up, operation, and shutdown of the unit are indicated, recorded, and controlled from the control room, into which access is supervised. The quantity and types of process instrumentation provided ensure the safe and orderly operation of all systems and processes over the full operating range of the station.

Reference chapters are as follows:

Title	Chapter
Engineered Safeguards	6
Instrumentation and Control	7

1.4.13 Fission Process Monitors and Controls

> Means are provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in the reactivity of the core.

Nuclear instrumentation is provided to monitor reactor power from the source range through the intermediate range and power range up to 120% of full power. The system provides indication, control, and alarm signals for reactor operation and protection.

The operational status of the reactor is monitored from the control room. When the reactor is subcritical, the relative reactivity status is continuously monitored and indicated by proportional counters located in instrument wells in the neutron shield tank adjacent to the reactor vessel. Two source detector channels supply information on multiplication while the reactor is subcritical.

When the reactor is critical, means for showing the relative reactivity status of the reactor are provided by control rod assembly bank positions displayed in the control room. The position of the control rod assembly banks is directly related to the reactivity status of the reactor when at power, and any unexpected change in the position of the control rod assembly banks under automatic control or any change in the coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. Periodic sampling to determine the boric acid concentration provides a long-term means of following reactivity status.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.14 Core Protection Systems

> Core protection systems, together with associated equipment, are designed to prevent or to suppress conditions that can result in exceeding acceptable fuel damage limits.

The reactor protection system receives, from unit instrumentation, signals that are indicative of an approach to an unsafe operating condition. This system then actuates alarms, prevents control rod assembly motion, initiates load runback, and/or opens the trip breakers causing the insertion of the control rod assemblies, depending on the severity of the condition. The allowable operating range within reactor trip settings includes combinations of power, temperature, and pressure that do not result in the occurrence of a departure from nucleate boiling with all reactor coolant pumps in operation.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.15 Engineered Safeguards Protection Systems

Protection systems are provided for sensing accident situations and initiating the operation of necessary engineered safeguards.

Instrumentation and controls provided for the protection systems are designed to trip the reactor when necessary to prevent or limit fission product release from the core and to limit energy release, to cause closure of containment isolation valves, and to control the operation of engineered safeguards equipment.

Additional tripping functions such as a high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water-level trip, loss-of-coolant-flow trip, steam and feedwater flow mismatch trip, steam generator low-low water-level trip, turbine trip, safety injection trip, neutron source and intermediate range trips, and manual trip are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

The passive accumulators of the safety injection system do not require signal or power sources to perform their function. The actuation of the active portion of this system is obtained from low-low pressurizer pressure, high containment pressure, steam header to steam line pressure differential, high steam flow coincident with a low T_{avg} or low steam line pressure signals, and manual actuation.

The containment isolation system provides the means for isolating various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a LOCA. The actuation of containment isolation is by coincident and redundant containment high-pressure signals.

Reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6
Instrumentation and Control	7

1.4.16 Monitoring Reactor Coolant Pressure Boundary

Means are provided for monitoring the reactor coolant pressure boundary to detect leakage.

Means of detecting leakage from the reactor coolant system are provided by measuring the airborne activity of the containment and indicating changes in makeup requirements and containment sump levels.

The sampling system for each unit contains two steam generator blowdown sample monitors in parallel. They are used for monitoring the liquid phase of the steam generators for radioactivity indicative of a primary-to-secondary system leak. Samples from each of the three steam generator bottoms are mixed in two common headers with each header going to a gamma scintillation counter mounted in an in-line liquid sampler. In general, both monitors are used continuously. Either monitor can be used to monitor any individual steam generator that is known to be leaking. In the event that one of the monitors becomes inoperative, or requires maintenance, the other monitor can be used to monitor any or all of the steam generators.

The output of the detectors is transmitted to the control room to provide indication, recording, and alarm functions. A high activity level is indicated by both audio and visual alarms.

The reference chapters are as follows:

Title	Chapter
Reactor Coolant System	4
Auxiliary and Emergency Systems	9
Radioactive Wastes and Radiation Protection	11

1.4.17 Monitoring Radioactive Releases

> Means are provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program is maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.

The containment atmosphere, the plant vent, and the waste disposal system liquid effluent discharge are monitored for radioactivity concentration during all normal operations, from anticipated transients, and from accident conditions.

All gaseous effluent from possible sources of accident releases of radioactivity external to the reactor containment (e.g., the spent-fuel pool and waste-handling equipment) are exhausted from monitored ventilation effluent pathways. Accident spills of liquids are maintained within the auxiliary building and collected in sumps. Any contaminated liquid effluent discharged to the condenser circulating water discharge canal is monitored. For the case of leakage from the reactor containment under accident conditions, the station radiation monitoring system, supplemented by portable survey equipment, will provide adequate monitoring of accident releases. The details of the procedures and equipment to be used in the event of an accident are specified in the station Emergency Plan Implementing Procedures (EPIPs).

The reference chapter is as follows:

Title	Chapter
Radioactive Wastes and Radiation Protection	11

1.4.18 Monitoring Fuel and Waste Storage

Monitoring and alarm instrumentation is provided for fuel and waste storage and handling areas for conditions that might contribute to a loss of continuity in decay heat removal and to radiation exposures.

The spent-fuel pool water temperature and level are continuously monitored. The temperature is displayed in the control room where an audible alarm sounds if the water temperature increases above a preset level. Audible alarms sound in the control room if the water level exceeds the high-level or low-level setpoints. The radiation level above the spent-fuel pool is continuously monitored by a radiation detector mounted on the fuel pool movable platform. A dose rate in excess of a preset level initiates an audible and visible alarm locally and in the control room. Continuous surveillance of radiation levels in the waste storage and handling areas is maintained by an appropriately mounted radiation detector. Radiation levels in excess of preset levels initiate audio and visual alarms locally and in the control room.

The reference chapters are as follows:

Title	Chapter
Auxiliary and Emergency Systems	9
Radioactive Wastes and Radiation Protection	11

1.4.19 Protection Systems Reliability

Protection systems are designed for high functional reliability and inservice testability necessary to avoid undue risk to the health and safety of the public.

The reactors use the Westinghouse magnetic-type control rod drive mechanisms that are similar to those used in the San Onofre, Indian Point, and Connecticut Yankee power stations. Upon a loss of power to the coils, the control rod assembly is released and falls by gravity into the core.

All reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and testing at power. The bypass removal of one trip circuit is accomplished by placing that circuit in a half-tripped mode; that is, a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel. Reliability and independence are obtained by redundancy within each tripping function.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Instrumentation and Control	7

1.4.20 Protection Systems Redundancy and Independence

Redundancy and independence designed into the protection systems are sufficient to ensure that no single failure or removal from service of any component or channel of such a system results in a loss of the protection function. The redundancy provided includes, as a minimum, two channels of protection for each protection function to be served.

The reactor protection system is designed in accordance with the IEEE *Standards for Nuclear Power Plant Protection Systems*.

Two reactor trip breakers are provided to interrupt power to the control rod drive mechanisms. The main breaker contacts are connected in series with the mechanism coils. Opening either breaker interrupts power to all mechanisms, causing them to release all control rod assemblies to fall by gravity into the core. Each breaker is opened through an undervoltage trip coil. A shunt trip relay is installed in parallel with the undervoltage attachment. Upon de-energization, contacts from the relay energize the reactor trip breaker shunt trip attachment and trips open the breaker. This provides a redundant/backup means to automatically trip the breakers upon the receipt of a trip signal from the reactor trip system. Each protection channel permits the actuation of one reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of power to the control rod drive mechanisms.

The initiation of the engineered safeguards provided for the LOCA is accomplished from redundant signals derived from reactor coolant system and containment instrumentation. Channel independence is carried throughout the system from the sensors to the output relays, including the power supplies for the channels.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.21 Single Failure Definition

Multiple failures resulting from a single event are treated as a single failure.

The requirements of this criterion are included in the criterion of Section 1.4.23.

1.4.22 Separation of Protection and Control Instrumentation Systems

> Protection systems shall be separated from control instrumentation systems to the extent that the failure or removal from service of any control instrumentation system component or channel, or of those components or channels common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

The coincident trip philosophy is employed to prevent a single failure from causing a spurious trip or from defeating the function of any channel.

Each reactor trip circuit is designed so that the trip occurs upon the de-energization of the circuit; an open circuit or loss of power to a channel will, therefore, cause that channel to go into its trip mode. Redundancy within each channel provides reliability and independence of operation. Channel independence is carried throughout the system from the sensor to the relay providing the logic. In some cases, however, it is desirable to employ a common sensor for both a control and a protection channel. Both functions are fully isolated in the remainder of the channel, control being derived from the primary safety signal path through an isolation amplifier. Thus, a failure in the control circuitry does not adversely affect the safety channel.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.23 Protection Against Multiple Disability for Protection Systems

> The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal or accident conditions, do not result in the loss of protection function or will be tolerable on some other basis.

The components of the reactor protection system are designed and arranged so that their environment in any emergency situation in which the components are required to function does not interfere with that function.

Each of the engineered safety features is designed to tolerate a single failure during the period of recovery following an incident, without loss of its protective function. This period of recovery consists of two segments, the short-term period and the long-term period. During the short-term period, the single failure is limited to a failure of an active component to complete its function as required. Should the single failure occur during the long-term rather than the short-term period, the safety-related system is designed to tolerate an active failure or a passive failure without loss of its protective function.

The following definitions pertain to the protection against multiple disability criteria:

Period of recovery - The time necessary to bring the plant to a cold shutdown and regain access to faulted equipment. The recovery period is the sum of the short- and long-term periods defined below.

Short term - The time from the initiation of the accident until the plant enters the recirculation phase of accident mitigation.

Long term - The time from when the plant enters the recirculation phase of the accident mitigation until the plant enters a cold shutdown mode and has the capability to access faulty equipment.

Active failure - The failure of a powered component, such as a piece of mechanical equipment, component of the electrical supply system, or instrumentation and control equipment, to act on command to perform its design function. Examples include the failure of a motor-operated valve to move to its correct position; the failure of an electrical breaker or relay to respond; the failure of a pump, fan, or diesel generator to start; etc.

Passive failure - The structural failure of a static component, which limits the component's effectiveness in carrying out its intended function. Examples include the failure of a battery or a cable.

Equipment moving spuriously from the proper safeguards position without signal, such as a motor-operated valve inadvertently shutting at the moment it is required, is not considered as an active failure.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.24 Emergency Power for Protection Systems

In the event of loss of all offsite power, sufficient alternative sources of power are provided to permit the required functioning of the protection systems.

There are four separate 120V ac vital buses, each supplied by an independent 15 kVA inverter power supply. The inverter is housed within an electrical cabinet, which also contains a rectifier/charger, a static transfer switch, a manual bypass switch, and a voltage regulating line conditioner (RLC). This configuration is shown in Reference Drawing 1. The inverters are supplied in pairs by a common station battery. Each inverter pair and one battery form a safety train of uninterruptable power. There are two station batteries and inverter pairs per nuclear unit at Surry, which provide two independent redundant uninterruptable power supply (UPS) electrical trains. Normally, the inverter load is absorbed by the UPS rectifier/charger. The emergency onsite power required to operate safety related protection systems equipment is supplied by three 100%-capacity diesel generators for the two units.

The reference chapter is as follows:

Title	Chapter
Electrical Systems	8

1.4.25 Demonstration of Functional Operability of Protection Systems

> Means shall be included for the suitable testing of the active components of protection systems while the reactor is in operation to determine if a failure or loss of redundancy has occurred.

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation. This includes a check through to the trip breakers that includes the trip logic. Thus, the operability of each trip channel is determined conveniently and without ambiguity.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.26 Protection Systems Fail-Safe Design

> The protection systems are designed to fail into the safe state or into a state established as tolerable on a defined basis if conditions such as a disconnection of the system, a loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized. An open circuit or loss of channel power therefore causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from two independent electrical buses. Failure to de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

The signal for containment isolation, except as initiated by safety injection, is developed from a three-out-of-four circuit in which each channel is separate and independent. The circuit signals for containment isolation upon high or high-high containment pressure. The failure of any one channel to energize when required does not interfere with the proper functioning of the isolation circuit. Each channel has provision for periodic tests to prove the ability to operate when energized.

Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on each drive, allowing the control rod assemblies to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The reference chapter is as follows:

Title	Chapter
Instrumentation and Control	7

1.4.27 Redundancy of Reactivity Control

Two independent control systems, preferably of different principles, are provided.

One of the two reactivity control systems employs control rod assemblies to regulate the position of neutron absorber within the reactor core. The other reactivity control system employs the chemical and volume control system to regulate the concentration of boron neutron absorber in the reactor coolant system.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Instrumentation and Control	7
Auxiliary and Emergency Systems	9

1.4.28 Reactivity Hot Shutdown Capability

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition.

The reactivity control systems are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for reload cores occurs at the beginning of life, no xenon conditions.

The control rod assemblies are divided into two categories, control groups and shutdown groups. The control groups, used in combination with soluble boron control, provide control of the reactivity changes of the core throughout the life of the core at power conditions. The control groups are used to compensate for short-term reactivity changes at power that might be produced by variations in reactor power requirements or in coolant temperature. The soluble boron control is used to compensate for the more slowly occurring changes in reactivity throughout core life as well as those attributable to fuel depletion and fission product buildup.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Auxiliary and Emergency Systems	9

1.4.29 Reactivity Shutdown Capability

One of the reactivity control systems provided is capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. The shutdown margin should ensure subcriticality with the most reactive control rod fully withdrawn.

The reactor core, together with the reactor control and protection system, is designed so that the minimum DNB ratio is at least the design DNBR limit (Section 3.2.3) and there is no fuel melting during normal operation, including periods of anticipated transients.

The shutdown groups of control rod assemblies are provided to supplement the control groups to make the reactor at least 1.77% delta k/k subcritical, following trip from any credible operating condition to the hot, zero-power condition, assuming the most reactive control rod assembly remains in the fully withdrawn position. Sufficient shutdown capability is also provided to ensure no DNB occurs for the most severe anticipated cooldown transient associated with a single active failure (i.e., the accidental opening of a steam bypass or relief valve). Thus, shutdown capability is achieved by a combination of control rod assemblies and automatic boron addition via the safety injection system with the most reactive control rod assembly assumed to be fully withdrawn. Manually controlled boron addition is used to supplement the control rod assemblies in maintaining the shutdown margin for the long-term conditions of xenon decay and unit cooldown.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Engineered Safeguards	6
Auxiliary and Emergency Systems	9

1.4.30 Reactivity Holddown Capability

The reactivity control systems provided are capable of (1) making the core subcritical under credible accident conditions, with appropriate margins for contingencies, and (2) limiting any subsequent return to power such that there is no undue risk to the health and safety of the public.

The reactivity control systems provided are capable of making and holding the core subcritical under accident conditions in a timely fashion, with appropriate margins for contingencies. Normal reactivity shutdown capability by control rod assemblies is provided within 2.4 seconds after a trip signal and this is followed by boron injection to compensate for the long-term xenon decay transient and for unit cooldown. Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boron required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Auxiliary and Emergency Systems	9

1.4.31 Reactivity Control System Malfunction

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

Reactor shutdown with control rod assemblies is completely independent of the normal control functions, since the trip breakers completely interrupt the power to the control rod drive mechanisms regardless of existing control signals. The protection systems limit reactivity transients so that the DNBR is not less than the design DNBR limit (Section 3.2.3) for any single malfunction in the reactor control system or in the de-boration controls.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Instrumentation and Control	7
Auxiliary and Emergency Systems	9

1.4.32 Maximum Reactivity Worth of Control Rods

Limits, which include reasonable margin, are placed on the maximum reactivity worth of control rods or elements and on the rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (1) rupture the reactor coolant pressure boundary, or (2) disrupt the core, its support structures, or other vessel internals sufficiently to lose the capability of cooling the core.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rod assemblies or elements and on the rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (1) rupture the reactor coolant pressure boundary or (2) disrupt the core, its support structures, or other vessel internals so as to lose the capability to cool the core.

The wiring arrangement for the control rod drive mechanisms prevents the withdrawal of control rod assemblies except as part of a select group of which each is part.

The maximum reactivity insertion rate is analyzed in a detailed unit analysis that assumes the two highest-worth sequential groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates that are well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No credible mechanical or electrical control system malfunction can cause a control rod assembly to be withdrawn at a speed greater than its mechanical limit.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Instrumentation and Control	7
Safety Analysis	14

1.4.33 Reactor Coolant Pressure Boundary Capability

> The reactor coolant pressure boundary is capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release is taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod drop, or cold water addition.

The reactor coolant pressure boundary is capable of accommodating without rupture the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a control rod assembly ejection.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod assemblies are used to control load variations only and core depletion is followed with boron dilution, only the control rod assemblies in the controlling group are inserted in the core at power, and these assemblies are only partially inserted. A control rod assembly insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

Through the arrangement of fuel assemblies, the design limits the maximum fuel temperature for the highest-worth ejected rod. This maximum temperature value precludes any resultant damage to the primary system pressure boundary such as gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a control rod to be rapidly ejected from the core is evaluated as a theoretical, though not a credible, accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the reactor coolant system and the reactor containment. The environmental consequences of rod ejection are less severe than those of the hypothetical loss of coolant, from which public health and safety are shown to be adequately protected.

The reference chapters are as follows:

Title	Chapter
Reactor	3
Reactor Coolant System	4
Safety Analysis	14

1.4.34 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

> The reactor coolant pressure boundary is designed and operated to reduce to an acceptable level the probability of a rapidly propagating failure. Consideration is given to (1) the provisions for control over service temperature and irradiation effects that may require operational restrictions, (2) the design and construction of the reactor pressure vessel in accordance with applicable codes, including those that establish the requirements for the absorption of energy within the elastic strain energy range and for the absorption of energy by plastic deformation, and (3) the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The reactor coolant pressure boundary is designed to reduce the probability of a rapidly propagating failure to an acceptable level.

The fast neutron exposure of the core region of the reactor vessel changes the notch toughness of the vessel material. This change is indicated by the increase in the nil ductility transition temperature and allowance for it is made in the operating procedures by ensuring that the vessel is not subjected to full operating pressure until its temperature exceeds the design transition temperature, defined to be the nil ductility transition temperature plus a 60°F margin. The pressure during unit start-up and shutdown at temperatures below the nil ductility transition temperature are maintained below the threshold of concern for safe operation.

The design transition temperature dictates the procedures to be followed in hydrostatic testing and in station operations to avoid excessive cold stress. The value of the design transition temperature is increased during the life of the station as required by the expected shift in the nil ductility transition temperature, which is confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the unit lifetime.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes.

The reference chapter is as follows:

Title	Chapter
Reactor Coolant System	4

1.4.35 Reactor Coolant Pressure Boundary Brittle Fracture Prevention

> For conditions under which reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation, or 60°F above the nil ductility temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Sufficient testing and analysis of materials used in reactor coolant system components are performed to ensure that the required nil ductility transition temperature limits specified in the criterion are met. Removable test capsules are installed in the reactor vessel and removed and tested at various times in the unit lifetime to determine the effects of the operation on system materials.

The reference chapter is as follows:

Title	Chapter
Reactor Coolant System	4

1.4.36 Reactor Coolant Pressure Boundary Surveillance

> Reactor coolant pressure boundary components have provisions for the inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming to current applicable codes is provided.

The design of the reactor vessel and its arrangement in the system permit accessibility during service life to all internal surfaces of the vessel and to certain external zones such as the areas of the nozzle-to-piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for the inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

The monitoring of the nil ductility transition temperature properties of the core region plates, forgings, weldments, and associated heat-treated zones is performed in accordance with ASTM E 185, *Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors*. Samples of reactor vessel plate materials are retained and cataloged in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile tests, but also tests of fracture mechanics specimens. The fracture mechanics specimens are the wedge-opening-loading-type specimens. The observed irradiation shifts in the nil ductility transition temperature of the core region materials are used to confirm the calculated limits to start-up and shutdown transients.

To define permissible operating conditions below the design transition temperature, a pressure range is established. The range is bounded by a lower limit for pump operation and an upper limit that satisfies reactor vessel stress criteria. To allow for thermal stresses during the heat-up or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of the rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected design transition temperature, brittle fracture during normal operation is not considered to be a credible mode of failure.

The reference chapter is as follows:

Title	Chapter
Reactor Coolant System	4

1.4.37 Engineered Safeguards Basis for Design

> Engineered safeguards are provided in the facility to back up the safety provided by the design of the core, the reactor coolant pressure boundary, and their protection systems. Such engineered safeguards are designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary and an unobstructed discharge from both ends.

Engineered safeguards are provided to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe in that boundary and an unobstructed discharge from both ends, and to separately cope with any steam or feedwater line break.

Limiting the release of fission products from the reactor fuel is accomplished by the safety injection system, which, by cooling the core, keeps the fuel in place and substantially intact and significantly limits the metal-water reaction.

A reinforced-concrete, steel-lined containment structure (Section 1.4.10), operating at subatmospheric pressure, is provided to enclose the entire reactor coolant system. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the reactor coolant system.

The reference chapter is as follows:

Title	Chapter
Engineered Safeguards	6

1.4.38 Reliability and Testability of Engineered Safeguards

> All engineered safeguards are designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

A comprehensive program of testing has been formulated for all equipment, systems, and system controls vital to the functioning of engineered safeguards. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the activation circuitry and mechanical components to ensure reliable performance, upon demand, throughout the unit lifetime.

The engineered safeguards components are checked periodically and routinely. In the event that one of the components requires maintenance as a result of failure to perform according to prescribed limits during the test, the necessary corrections or minor maintenance are accomplished and the component is retested immediately.

The reference chapter is as follows:

Title	Chapter
Engineered Safeguards	6

1.4.39 Emergency Power for Engineered Safeguards

> Alternative power systems are provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning required of the engineered safeguards. As a minimum, the onsite power system and the offsite power system each, independently, provide this capacity, assuming the failure of a single active component in each power system.

Two independent sections of emergency 4160V buses and switchgear are provided for each unit. Each section is sized to carry 100% of the emergency load and may be energized by either onsite or offsite power supplies. The onsite and offsite power supplies are both independently capable of supplying power to the engineered safeguards. This capability is maintained even in the event of a failure of any single active component in either system. In the unlikely event of total loss of offsite power, the emergency 4160V buses are energized by the emergency diesel generators. Three diesel generators are available for two units. One diesel is exclusively for Unit 1, the second is exclusively for Unit 2, and the third functions as a backup for either unit. Each diesel generator is connected to one of the emergency buses, and each bus is connected to one set of the duplicated engineered safeguards equipment, thus ensuring operations of safeguards equipment under all conditions, including the failure of a single component in each power system. Sections 8.4.1 and 8.5 discuss the alternate station power systems and emergency power system, respectively.

Tests of the automatic operation of the power source transfer system at the 4160V level are made during shutdown for refueling to ensure that station on-site power is supplied automatically when an offsite power source is out of service. The periodic starting and loading of each emergency diesel generator and its emergency bus ensures the operability of the emergency power supply in the event of loss of off-site power.

The reference chapter is as follows:

Title	Chapter
Electrical Systems	8

1.4.40 Missile Protection

> Protection for engineered safeguards is provided against dynamic effects and missiles that might result from plant equipment failures.

Layout and structural design specifically protects the injection lines leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant system pipe rupture. The separation of individual injection lines is provided to the maximum extent practicable. The movement of injection lines associated with the rupture of a reactor coolant loop is accommodated by line flexibility and by the design of the pipe supports, so that no damage beyond the missile barrier is credible.

The reference chapters are as follows:

Title	Chapter
Reactor Coolant System	4
Containment System	5
Engineered Safeguards	6

1.4.41 Engineered Safeguards Performance Capability

> Engineered safeguards, such as the safety injection system and the containment heat removal system, provide sufficient performance capability to accommodate the failure of any single active component without any undue risk to the health and safety of the public.

The overall capability of the engineered safeguards meets the suggested requirements of 10 CFR 50.67 or RG 1.183, as applicable, for the occurrence of any rupture of a reactor coolant or main steam system pipe, including the double-ended rupture of a reactor coolant pipe, known as the design-basis accident.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6
Safety Analysis	14

1.4.42 Engineered Safeguards Components Capability

Engineered safeguards are designed so that the capability of these features to perform their required function is not impaired by the effects of a LOCA to the extent of causing undue risk to the health and safety of the public.

Instrumentation, motors, cables, and penetrations inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The safety injection system piping serving each loop is anchored at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force of any branch line served from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equivalent to that which produces failure of the piping under the action of free and unrestrained discharge to atmosphere or the motion of the broken reactor coolant pipe to which the safety injection system pipes are connected. This prevents possible failure at any point upstream from the support point, including the branch line connection into the piping header.

The containment spray and recirculation spray piping has been installed with sufficient anchors, constraints, and guides to withstand the effects of operating-basis and design-basis earthquakes. This piping has also been installed to withstand the effects of dead weight, thermal, and pressure forces in the piping. Redundant containment spray and recirculation spray piping systems have been installed to preclude the possibility of sprays being lost as a result of pipe failure.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6

1.4.43 Accident Aggravation Prevention

Protection against any action of the engineered safeguards that accentuates significantly the adverse after effects of a loss of normal cooling is provided.

The reactor is maintained subcritical following a pipe rupture accident. The introduction of borated cooling water into the core does not result in a net positive reactivity addition. The control rod assemblies insert and remain inserted.

The supply of water by the safety injection system to cool hot core cladding does not produce significant metal-water reactions.

The delivery of cold emergency core cooling water to the reactor vessel following accidental expulsion of reactor coolant does not cause a further loss of integrity of the reactor coolant system boundary.

The reference chapter is as follows:

Title	Chapter
Engineered Safeguards	6

1.4.44 Safety Injection System Capability

> A safety injection system with the capability for accomplishing adequate emergency core cooling is provided. This core cooling system and the core are designed to prevent fuel and clad damage that interferes with the emergency core cooling function and to keep the clad metal-water reaction within acceptable limits for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such a safety injection system is evaluated conservatively in each area of uncertainty.

The safety injection system employs a passive system of accumulators that do not require any external signals or source of power for their operation to cope with the short-term cooling requirements of a large reactor coolant pipe break. The high-head and the low-head safety injection systems, each capable of supplying the required emergency cooling, are also provided for small-break protection and to keep the core submerged after the accumulators have discharged following a large break. These systems are arranged so that the single failure of any active component does not interfere with meeting the short-term cooling requirements.

The high-head and low-head safety injection systems are each capable of fulfilling long-term cooling requirements. The failure of any single active component or the development of excessive leakage during the long-term cooling period does not interfere with the ability to meet necessary long-term cooling objectives with one of the systems.

The primary purpose of the safety injection system is to automatically deliver cooling water to the reactor core in the event of a LOCA. This limits the fuel clad temperature and thereby ensures that the core remains intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

2. A loss of coolant associated with the rod ejection accident.
3. A steam generator tube rupture.

The basic design criteria for LOCA evaluations are:

1. The cladding temperature is less than
   a. The melting temperature of zirconium alloy cladding material.
   b. The temperature at which gross core geometry distortion, including clad fragmentation, may be expected.
2. The total core metal-water reaction is limited to less than 1%.

Meeting these criteria ensures that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the safety injection system adds shutdown reactivity so that with an assumed stuck control rod assembly, no offsite power, and minimum engineered safeguards, there is no consequential damage to the primary system, and the core remains in place and intact. When there is no stuck control rod assembly, offsite power is available, and all equipment is operating at design capacity, there is no significant cladding rupture.

The reference chapters are as follows:

Title	Chapter
Engineered Safeguards	6
Safety Analysis	14

1.4.45 Inspection of Safety Injection System

> Design provisions, where practical, are made to facilitate the physical inspection of all critical parts of the safety injection system, including reactor vessel internals and water injection nozzles**.**

Design provisions are made for the inspection of all components of the safety injection system to the extent practical**.** An inspection is performed periodically to demonstrate system readiness.

The pressure containment boundaries can be inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

In addition, critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps can be inspected visually or by boroscopic examination for evidence of erosion, corrosion, and vibration wear, and non-destructive tests can be performed where such techniques desirable, practical, and appropriate.

The reference chapter is as follow:

Title	Chapter
Engineered Safeguards	6

1.4.46 Testing of Safety Injection System Components

> Design provisions are made so that components of the safety injection system can be tested periodically for operability and functional performance.

The design provides for the periodic testing of active components of the safety injection system for operability and functional performance.

Preoperational performance tests of the components were performed in the manufacturer's shop. An initial system flow test, performed prior to initial criticality, demonstrated the proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Each active component of the safety injection system may be individually actuated on the normal power source at any time during station operation to demonstrate operability. The test of the safety injection pumps, which perform as charging pumps during normal operation, employs a minimum-flow recirculation test line that connects back to the volume control tank. Remotely operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers also may be checked during integrated system test periods.

The reference chapters are as follows:

Title	Chapter
Engineered Safeguards	6
Instrumentation and Control	7

1.4.47 Testing of Safety Injection System

> Capability is provided to test periodically the operability of the safety injection system up to a location as close to the core as is practical.

Design provisions include special instrumentation, testing, and sampling lines to perform the tests, and unit shutdown to demonstrate the proper automatic operation of the safety injection system. A test signal is supplied to initiate automatic action. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. In addition, other tests are performed periodically to verify that the safety injection pumps attain required discharge heads.

The accumulator tank pressure and level are continuously monitored during unit operation, and flow from the tanks can be checked at any time using test lines.

The reference chapters are as follows:

Title	Chapter
Engineered Safeguards	6
Instrumentation and Control	7

1.4.48 Testing of Operational Sequence of Safety Injection System

> Capability shall be provided to test, under conditions as close as practical to design, the full operational sequence that would bring the safety injection system into action, including the transfer to alternative power sources.

The design provides for the capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. This functional test is performed with the reactor coolant system initially cold and at low pressure. The safety injection system valving is set to initially simulate the system alignment for power operation. This test may be conducted on the normal shutdown power system, and it may include transfer to the alternative power source.

During the initial system checkout, the functioning of the accumulators is checked by closing the stop valve, raising the pressure in the tank, and then opening the stop valve and observing the rising pressurizer level. The rising water level in the pressurizer provides an indication of system delivery.

The reference chapter is as follows:

Title	Chapter
Engineered Safeguards	6

1.4.49 Containment Design Basis

> The containment structure, including access openings and penetrations and any necessary containment heat removal systems, is designed to accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including a considerable margin for the effects of metal-water or other chemical reactions that can occur as a consequence of the failure of safety injection systems.

The design of the containment structure is based on the design basis accident, discussed in Section 14.5.1, which assumes a double-ended rupture of the largest pipe in the reactor coolant system, coupled with partial loss of the redundant engineered safeguards systems (minimum safeguards). The maximum containment pressure reached in a design basis accident is less than the 45-psig design limit. Further, the containment analyses performed assume a 2% metal-water reaction which is well above the less than 1% expected for all accidents considered.

The containment structure, including access openings and penetrations, is designed to withstand a pressure of 45 psig and the associated thermal effects without exceeding the design leakage rate of 0.1 weight percent of containment air per 24 hours.

The heat removal capacity of the containment spray systems for the minimum safeguards returns the containment pressure to a subatmospheric condition in less than 60 minutes after a design-basis accident. This original design criterion was modified in conjunction with the analyses for implementation of the alternative source term. The modified criteria require that, following the LOCA, the containment pressure be less than 0.5 psig for Unit 1 (1.0 psig for Unit 2) within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for Unit 1 (1.0 psig for Unit 2) for the interval from 1 to 4 hours following the Design Basis Accident. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6
Safety Analysis	14

1.4.50 Nil Ductility Transition Temperature Requirement for Containment Material

Principal load-carrying components of ferritic materials exposed to the external environment are selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

The containment liner is not exposed to the external environment. However, the containment liner has sufficient ductility to tolerate local deformations without rupture. The liner material has a nil ductility transition temperature of -20°F, which is 80°F below the normal minimum shutdown temperature of 60°F. The equipment and personnel hatches are made of steel with a nil ductility transition temperature of -20°F. Exposed hatch surfaces during station operation are not expected to be at a temperature below 10°F.

The reference chapter is as follows:

Title	Chapter
Containment System	5

1.4.51 Reactor Coolant Pressure Boundary Outside Containment

If part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, are provided to protect the health and safety of

> the public in case of an accidental rupture in that part. The determination of the appropriateness of features, such as isolation valves and additional containment, includes a consideration of the environmental and population conditions surrounding the site.

No portions of the reactor coolant pressure boundary extend beyond the containment barrier.

1.4.52 Containment Heat Removal Systems

> Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, are provided.

Four separate containment recirculation spray subsystems, each with approximately 50% capacity, serve to remove heat from the containment after a LOCA, as described in Section 6.3.1. Each subsystem contains one deepwell-type pump. In two subsystems, the recirculation spray pumps are located inside the containment. In the other two subsystems, the recirculation spray pumps are located in the containment auxiliary structures and are accessible for servicing at all times.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6

1.4.53 Containment Isolation Valves

> Penetrations that require closure for the containment function are protected by redundant valving and associated apparatuses.

All penetrations requiring valve closure for containment isolation have redundant valving so that the failure of one valve does not prevent the isolation of the containment. No manual operation or action is required to activate the valves to effect isolation. All remotely actuated valves have their positions indicated in the control room at group visual position indicators.

The reference chapter is as follows:

Title	Chapter
Containment System	5

1.4.54 Initial Containment Leakage Rate Testing

> The containment is designed so that integrated leakage rate testing can be conducted at design pressure after the completion and installation of all penetrations, and the leakage rate can be measured over a sufficient period of time to verify its conformance with required performance.

Refer to the response to the criterion in Section 1.4.55.

1.4.55 Periodic Containment Leakage Rate Testing

> The containment is designed so that integrated leakage rate testing can be done periodically at design pressure during the plant lifetime.

The test frequency, test pressure, and type of test used are in accordance with Technical Specifications.

The completed containment structure, with all necessary penetrations, is designed so that leakage does not exceed 0.1% of the contained volume per day at the design pressure of 45 psig. Upon completion of the construction of the containment structure and the installation of all penetrations, Type A tests of the containments were performed at 39.2 psig and 25 psig. The tests were performed using the leakage monitoring system described in Section 5.3.2. Since the normal operating pressure of the containment is subatmospheric, containment leakage is monitored continuously by means of the leakage monitoring system.

The periodic leakage rate retest is conducted at a single test pressure. During the interval between the periodic leakage rate retests, a series of periodic surveillance tests (Type B and C tests) are carried out to monitor the principal sources of leak development.

The reference chapter is as follows:

Title	Chapter
Containment System	5

1.4.56 Provision for Testing of Penetrations

> Provisions are made for testing penetrations that have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

All penetrations having resilient seals or expansion bellows are fitted with test connections to permit pressurization to 50 psig to demonstrate leaktightness.

The reference chapter is as follows:

Title	Chapter
Containment System	5

1.4.57 Provision for Testing of Isolation Valves

> The capability is provided for testing the functional operability of valves and associated apparatuses essential to the containment function, for establishing that no failure has occurred, and for determining that valve leakage does not exceed acceptable limits.

Type C tests are performed on the isolation valves to verify their sealing capability and leaktightness as described in Section 5.5. The tests include valve closure and leakage tests. Isolation valves, which are normally closed, are exercised to verify closure and sealing capabilities. Valve leakage tests are performed in accordance with the requirements of the Type C test.

The reference chapter is as follows:

Title	Chapter
Containment System	5

1.4.58 Inspection of Containment Pressure-Reducing Systems

Design provisions are made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems such as pumps, valves, spray nozzles, torus, and sumps.

Equipment composing the engineered safeguards systems is so situated that periodic physical inspections can be made. All equipment can be inspected during planned refueling shutdowns.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6

1.4.59 Testing of Containment Pressure-Reducing Systems Components

The containment pressure-reducing systems are designed so that active components such as pumps and valves can be tested periodically for operability and required functional performance.

The containment recirculation spray pumps and valves are tested, periodically, by manually closing the required breakers in the control room to test actuation and component operation. Bypass lines on the recirculation spray pumps located outside the containment permit flow measurements to be made, which can then be compared to the results of preoperational tests. The recirculation spray pumps located inside the containment are periodically tested to ensure their operability.

Bypass lines to the refueling water storage tank permit brief operational tests of the containment spray pumps. Periodic tests of the CS pump discharge MOVs demonstrate that they are functioning properly. Test air connections on the containment spray discharge lines, installed prior to the nozzle air tests, ensure that these lines and the check valves are open.

The reference chapter is as follows:

Title	Chapter
Engineered Safeguards	6

1.4.60 Testing of Containment Spray Systems

A capability is provided to test periodically the delivery capability of the containment spray systems at a position as close to the spray nozzles as is practical.

Provision is made to permit the testing of the containment spray system and the containment recirculation spray system throughout the life of the unit to ensure that the systems are operational. For preoperational testing, the ends of the spray headers are fitted with blind flanges that allow the connection of temporary drain lines for full-flow testing up to the nozzles. Such testing allows for the testing of the spray systems over the full range of flow and starting conditions.

The reference chapter is as follows:

Title	Chapter
Engineered Safeguards	6

1.4.61 Testing of Operational Sequence of Containment Pressure-Reducing Systems

A capability is provided to test, under conditions as close to design considerations as practical, the full operational sequence that brings the containment pressure-reducing systems into action, including the transfer to alternative power sources.

The design of the control system for the containment spray system and the containment recirculation spray system includes manual test switches that provide for the individual testing of all the equipment in the systems and the testing of the operational sequence of the spray systems. These tests may be conducted on the normal shutdown power system or an alternative power source.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Engineered Safeguards	6
Instrumentation and Control	7

1.4.62 Inspection of Air Cleanup Systems

Design provisions are made to facilitate the physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers.

Refer to the response to the criterion in Section 1.4.65.

1.4.63 Testing of Air Cleanup Systems Components

> Design provisions are made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Refer to the response to the criterion in Section 1.4.65.

1.4.64 Testing of Air Cleanup Systems

> A capability is provided for the in situ periodic testing and surveillance of the air cleanup systems to ensure that (1) filter bypass paths have not developed and (2) filter and trapping materials have not deteriorated beyond acceptable limits.

Refer to the response to the criterion in Section 1.4.65.

1.4.65 Testing of Operational Sequence of Air Cleanup Systems

> A capability is provided to test, under conditions as close to design conditions as practical, the full operational sequence that brings the air cleanup systems into action, including the transfer to alternative power sources and the design air flow delivery capability.

Engineered safeguards for the Surry Power Station do not include a postaccident air cleanup system. The containment ventilation system is normally in continuous service and is equipped to handle activity associated with normal station operation. No special tests or inspections of this system are performed.

1.4.66 Prevention of Fuel Storage Criticality

> Criticality in the new-fuel and spent-fuel storage areas is prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration of the reactor coolant system and the fuel transfer canal, reactor cavity, and spent-fuel pool is maintained at not less than that required to shut down the core to a $k_{eff} = 0.95$ with all control rods inserted. This concentration is sufficient to ensure that $k_{eff} < 1.00$ even if all control rods are withdrawn.

The new-fuel storage racks are designed so that it is impossible to insert assemblies in violation of the design in other than the lattice spacing. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to ensure an ever-safe geometry.

The spent-fuel storage racks are designed so that it is impossible to insert assemblies in violation of the design in other than the lattice spacing. Borated water is used to fill the spent-fuel pool at a concentration to match that used in the reactor cavity during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to ensure $k_{eff} \leq 0.95$, even if unborated water is used to fill the pool.

The fuel transfer equipment is designed to handle one fuel assembly at a time. The new-fuel storage area cannot be flooded.

The reference chapter is as follows:

Title	Chapter
Auxiliary and Emergency Systems	9

1.4.67 Fuel and Waste Storage Decay Heat

> Reliable decay heat removal systems are designed to prevent damage to the fuel in storage facilities that can result in radioactivity release to plant-operating areas or the public environs.

Decay heat from spent fuel is dissipated in the water of the spent-fuel pool and subsequently removed by a cooling system. Redundancy of system components is provided to ensure the maintenance of storage pool water cleanliness and level, and to remove heat from the water.

The reference chapter is as follows:

Title	Chapter
Auxiliary and Emergency Systems	9

1.4.68 Fuel and Waste Storage Radiation Shielding

> Shielding for radiation protection is provided in the design of fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

The spent-fuel storage pool is designed to meet 10 CFR 20 requirements in providing radiation shielding for operating personnel during fuel transfer and during storage of spent fuel. Work areas adjacent to the canal wall are shielded; however, barricades are necessary to limit personnel access during actual fuel transfers.

Waste storage and processing facilities in the auxiliary building area have shielding meeting 10 CFR 20 requirements for operating personnel.

Periodic surveys by health physics personnel using portable radiation detectors ensure that radiation design levels are not degraded during unit lifetime.

The reference chapters are as follows:

Title	Chapter
Auxiliary and Emergency Systems	9
Radioactive Wastes and Radiation Protection	11

1.4.69 Protection Against Radioactivity Release From Spent Fuel

The containment of fuel and waste storage is provided if accidents could lead to the release of undue amounts of radioactivity to the public environs.

Spent fuel systems are designed to preclude gross mechanical failures that could lead to significant radioactivity releases. In addition, during refueling, fuel building ventilation air is passed through charcoal filters, containment ventilation air is monitored, and, if airborne radioactivity increases beyond a predetermined value, the containment ventilation system is isolated automatically.

Liquid waste storage facilities are designed so that any possible release of waste liquids is contained within the facility and does not result in an uncontrolled release to the environment. Any waste liquid leakage or release from components within the auxiliary building, fuel building, decontamination building, or radwaste facility flows directly to the vent and drain system or is collected in sumps and pumped to the liquid waste disposal system. The boron recovery tanks located in the station yard area are in separately diked, Class I structures, each of which is of sufficient capacity to retain the total liquid volume resulting from the rupture of one of these tanks. Radioactive gases are stripped from the liquid stored in the boron recovery tanks so that a tank failure does not constitute a significant gaseous release.

Waste gas inventories are carefully monitored and controlled so that no single component failure would result in a whole-body dose at the site boundary greater than 25 rem. All gaseous discharges from the station are continuously monitored for particulate and gaseous radioactivity during the release.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Auxiliary and Emergency Systems	9
Radioactive Wastes and Radiation Protection	11
Safety Analysis	14

1.4.70 Control of Releases of Radioactivity to the Environment

The facility design includes those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity is provided for the retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control is justified (1) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (2) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence, except that reductions of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

The control of waste gas effluents is accomplished by the holdup of waste gases in buried, double-wall decay tanks until the activity of tank contents and existing environmental conditions permit discharges within 10 CFR 20 requirements. In addition, waste gas effluents are monitored at the point of discharge for radioactivity and rate of flow. No decay tank failure results in an activity release greater than 10 CFR 100 limits.

The control of liquid waste effluents is maintained by batch processing all liquids, sampling them before discharge, and controlling their rate of release, and by preventing inadvertent tank discharges. Liquid effluents are monitored for radioactivity and rate of flow. Liquid waste disposal system collection and surge tank, and the evaporator, reverse osmosis, and ion exchange capacities are sufficient to handle any expected transient in the development of liquid waste volume.

Station solid wastes are prepared batchwise for offsite disposal by approved contractors. Solid wastes are prepared for shipment by placement in shielded and reinforced containers that meet regulatory requirements.

The reference chapters are as follows:

Title	Chapter
Containment System	5
Auxiliary and Emergency Systems	9
Radioactive Wastes and Radiation Protection	11
Safety Analysis	14

1.4 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11448-FE-1A2	One Line Integrated Schematic, Electrical Power Distribution, Units 1 & 2

1.5 COMMON FACILITIES

Separate and similar systems and equipment are provided for each unit, except as noted below. Where some components of a system are shared by both units, only those components that are shared are listed.

1. Electrical systems (Section 8.2)

   Standby station service transformer

   Backup emergency diesel generator

2. Chemical and volume control system (Section 9.1)

   Chemical-mixing tank

   Boric acid storage tanks (three)

   Boric acid pumps (four)

   Batching tank

   Resin fill tank

3. Boron recovery system (Section 9.2)
4. Component cooling water system (Section 9.4)

   Component-cooling surge tank

   Component-cooling pumps (four)

   Component-cooling heat exchangers (four)

5. Fuel pool cooling system (Section 9.5)

   Fuel pool circulation pumps (two)

   Fuel pool skimmer pumps (two)

   Fuel pool coolers (two)

   Fuel pool skimmer filters (two)

   Fuel pool purification filter (one)

   Fuel pool ion exchanger

   Fuel pool purification pumps (two)

6. Sampling system (Section 9.6)

7. Vent and drain system (Section 9.7)

   Auxiliary building sump pumps (two)

   Fuel building sump pumps (two)

   Liquid waste strainers (two)

8. Service water system (Section 9.9)
9. Fire protection system (Section 9.10)
10. Ventilation system (Section 9.13) (other than containment ventilation)
11. Heating boilers (Section 10.3.2)
12. Lubricating oil system (Section 10.3.7)

    Clean and dirty lube-oil storage tanks (two)

    Transfer pump

13. Radioactive waste systems (Section 11.2)
14. Structures, buildings, and miscellaneous

    Auxiliary building

    Fuel building

    Turbine building and turbine room crane

    Service building

    Main control area

    Decontamination building

    Office building

    General station services, nonelectrical

    Fuel-oil system

    Service water pump house

    Fire-pump house

    Intake and discharge canals

    Screenwell

    Laundry facility

    Radwaste facility

1.6 RESEARCH AND DEVELOPMENT REQUIREMENTS

(Note: This is the initial plant research and development. Any post research and development is described in the individual sections.)

The design is based on proven concepts that have been developed and successfully applied to the design of PWR systems. Results of work completed under the Nuclear Safety Research and Development Program conducted by the AEC were incorporated in the design and evaluation of applicable portions of the engineered safety features.

The term "research and development" as used in this section is the same as that used by the Commission in Section 5.2 of its regulations, as follows:

> (n) "Research and development" means (1) theoretical analysis, exploration or experimentation; or (2) the extension of investigative findings and theories of the scientific nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, materials and processes.

The research and development discussed in the FSAR is to confirm the engineering and design values normally used to complete equipment and system designs. It does not involve the creation of new concepts or ideas.

The technical information generated is used either to demonstrate the safety of the design and more sharply define margins of conservatism, or to lead to design improvements.

The schedules for development of this technical information were compatible with the plant schedule such that definite results were made available before the plant design was complete.

The Westinghouse research and development programs under way during the FSAR stages of the Surry project are listed in WCAP-7498-L, *Topical Report, Safety Related Research and Development for Westinghouse Pressurized Water Reactors*, Spring 1970 (Reference 1). This topical report is upgraded periodically.

The specific areas in which additional information was developed and which were required for unit operation are as follows:

1. Core stability evaluation.
2. Fuel rod burst program.

Other areas of research and development are those that gave added confirmation that the overall design was conservative. These programs were carried out basically to provide technical information that could be applied to component or system optimization in future plants. These programs included the following:

1. Burnable poison program.

2. Blowdown forces program.
3. Reactor vessel thermal shock analysis program.
4. Containment spray program.
5. Fuel development program.
6. Incore detector program.
7. Empire States Atomic Development Associates DNB program.
8. Full Length Emergency Core Cooling Heat Transfer Test program.
9. Flashing heat transfer program.
10. Loss of coolant analysis program.

These programs are discussed extensively in WCAP-7498-L (Reference 1).

1.6.1 Required Research and Development

There are two programs which were required for plant operation: the core stability evaluation and fuel rod burst programs.

1.6.1.1 Core Stability Evaluation Program (Item 1 of Reference 1)

The purpose of this program was to establish means for the detection and control of potential xenon oscillations and for the shaping of the axial power distribution for improved core performance. This program was completed in two areas:

1. Confirmation of the ability of the ex-core detector system to indicate gross core power distribution sufficient to permit xenon oscillation within specified operating limits.
2. Development of a control system using the ex-core detector system and part-length control rods. (It should be noted that the part-length rods were removed by a design change initiated in 1978.)

The third part of this program, verification through start-up testing that the control system can control the core power distribution and that adequate margins exist to operate the Surry unit, was carried out on Westinghouse reactors that were placed in operation before Surry. These included H. B. Robinson Unit 2 (Docket 50-261) and Turkey Point Unit 3 (Docket 50-250).

1.6.1.2 Fuel Rod Burst Program (Item 2 of Reference 1)

The basic design criteria for LOCA evaluations are given in Section 14.5.

Satisfaction of these criteria ensures that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

The effect of rod bursting, swelling, or shattering must be considered in the loss-of-coolant evaluations. In the blowdown phase of the accident, core geometry distortion may result from clad bursting or swelling. The clad temperature may get sufficiently high (1200° to 2000°F) that a bursting or swelling of the clad would occur by virtue of the internal gas pressure and a significant reduction of clad strength. Clad bursting or swelling is of concern because of the possibility of blocking the flow channel so that coolant flow would be insufficient to meet the above LOCA design criteria.

A program to investigate the performance of fuel rods during a simulated LOCA was completed. It supplied empirical data on the above safety-related problems from which the amount and kinds of geometry distortion can be predicted over the range of conditions of interest. The effects of this geometry distortion on the ability of the emergency core cooling system to meet the LOCA design criteria were determined using analytical design techniques.

1.6.1.2.1 Single-Rod Burst Tests (SRBT)

The performance of the fuel rods during a simulated LOCA was evaluated in a test program that is described in WCAP-7379-L, Volume I and Volume II (Reference 2).

Volume I (Westinghouse Proprietary) describes burst, quench, and eutectic formation tests with unirradiated tubes and provides an evaluation of the data from both reports. An interpretation with regard to the postulated sequence during the LOCA is given.

Volume II (Non-proprietary) reports the results of work under AEC Contract AT-(30-1)-3017 and describes burst and quench tests on irradiated tubes.

The single-rod tests indicated that rod-to-rod interference might occur following rod burst and must be considered. The quantitative evaluation of the influence of adjacent rods in a fuel assembly would be difficult, if not impossible, to determine analytically. Therefore, the rod burst program was extended to include multi-rod burst tests. Multi-rod burst tests (MRBT) were performed to demonstrate that the rods in a PWR rod bundle burst randomly so that a minimal-flow channel area, for core-cooling purposes, is maintained.

1.6.1.2.2 Multi-Rod Burst Test

The results of this phase of the rod burst program are reported in WCAP-7495-L, Volume I and Volume II (Westinghouse Proprietary) (Reference 3).

Volume I describes the test apparatus and conditions and provides an evaluation of the test results. Volume II presents the application of the MRBT results to the LOCA core thermal analysis.

The MRBT results show that the burst locations are staggered axially along the fuel rods and that, to some degree, rod-to-rod contact does occur. However, the remaining flow area is always sufficient to ensure adequate core cooling. Analytical evaluations of a typical double-ended cold-leg break, considering flow redistribution due to the geometry distortion and rod-to-rod contact, have shown that the peak clad temperature increases approximately 70°F over the 2300°F peak temperature without geometry distortion.

The program was completed and results were satisfactory. No backup research and development measures were considered necessary.

1.6.2 Other Research and Development

Other areas of research and development included those described below.

1.6.2.1 Burnable Poison Program (Item 7 of Reference 1)

Burnable poison rod development is complete. The burnable poison rods are borosilicate glass encased in stainless steel tubes. The fixed rods are used to reduce the concentration of boric acid poison in the moderator, thus ensuring that the moderator coefficient of reactivity is always negative at operating temperature.

1.6.2.2 Blowdown Forces Program (Item 15 of Reference 1)

The objective of the program was to develop digital computer programs for the calculation of pressure, velocity, and force transients in the reactor coolant system during a LOCA, and to use these codes in the calculation of blowdown forces on the fuel assemblies and reactor internals to ensure that the stress and deflection criteria used in the design of these components are met.

Westinghouse completed the development of BLODWN-2, an improved digital computer program for the calculation of local fluid pressure, flow, and density transients in the primary coolant system.

Extensive comparisons were made between BLODWN-2 and test data, and the results are given in WCAP-7401 (Reference 4). Agreement between code predictions and data was good.

An analysis using the BLODWN-2 program was completed for the Indian Point Unit 2 reactor. It was concluded from the analysis that the design of this reactor met the established design criteria. Designs for subsequent Westinghouse pressurized water reactors included the use of the BLODWN-2 program.

1.6.2.3 Reactor Vessel Thermal Shock (Item 16 of Reference 1)

The effects of safety injection water on the integrity of the reactor vessel following a postulated LOCA were analyzed using data on the fracture toughness of heavy section steel, both at beginning of plant life and after irradiation, corresponding to approximately 40 years of equivalent plant life. The results showed that, under the postulated accident conditions, the integrity of the reactor vessel is maintained.

Fracture toughness data were obtained from a Westinghouse experimental program associated with the heavy section steel technology (HSST) program at the Oak Ridge National Laboratory and with the Euratom programs. Since results of the analyses were dependent on the fracture toughness of irradiated steel, efforts continued to obtain additional fracture toughness data. The HSST program was scheduled for completion by 1973.

A detailed analysis (Reference 5) of the linear elastic fracture mechanism method, along with various sensitivity studies, was submitted to the AEC staff and members of the Advisory Committee on Reactor Safety.

Revised material for this report, plus additional analytical and fracture toughness data, was presented at a meeting with the Containment and Component Technology Branch on August 9, 1968, and forwarded by letter for AEC review and comment on October 29, 1968.

It was not anticipated that the HSST program would lead to any new conclusions about the Surry reactor vessel integrity under LOCA conditions.

Several backup positions are available if vessel integrity cannot be ensured for the full plant life with the operating modes presently used. one solution would be to anneal the reactor vessel such that material properties approach their original values. This solution is feasible, in principle, and could be performed with the vessel in place.

Note: Refer to Section 18.3.1.2 regarding the 20-year period of extended operation beyond the original 40-year operating license.

1.6.2.4 **Containment Spray Program (Item 3 of Reference 1)**

In the unlikely event of a major LOCA, one of the radiological hazards could be the release into the containment of radioactive iodine from ruptured fuel. The absorption of this iodine by a suitable chemical spray has been investigated extensively by Vepco and Westinghouse. The research and development program is discussed in WCAP-7499-L (Reference 6).

1.6.2.5 **Fuels Development Program for Operation at High Power Densities (Item 8 of Reference 1)**

As part of the program to demonstrate the satisfactory operation of fuel at high burnup and power densities, fuel was tested in both the Saxton and Jose Cabrera (Zorita, Spain) reactors. The Saxton loose-lattice irradiation program was used to demonstrate fuel performance at conditions significantly in excess of 1970 PWR design limits, and to establish power burnup limits for the fuel. The Jose Cabrera reactor was the first PWR with a Zircaloy core to operate at similar core conditions to the 1970 design units. Because of the timely manner in which fuel could be irradiated in Jose Cabrera, four fuel assemblies were tested there to demonstrate the satisfactory operation of the fuel in a commercial PWR environment.

The sustained successful operation of special Jose Cabrera fuel rods at peak design power levels (in excess of those planned for these units) also increased the assurance that the fuel had adequate performance margins to accommodate transient overpower operation.

1.6.2.6 Incore Detector Program (Item 9 of Reference 1)

The purpose of this program was to develop fixed incore neutron detectors suitable for the continuous monitoring of the power distribution in a PWR core.

Testing at San Onofre, the Western New York Research Reactor, the Brookhaven high flux beam reactor, and the Union Carbide (Tuxedo) reactor were used to evaluate detector performance. Tests at the Tuxedo reactor were performed for detector linearity and the optimization of design. Cables for incore detectors were also tested. Cable reliability was greatly improved in this program.

This program permitted a fixed incore flux detector system to be installed in H. B. Robinson Unit 2 and showed the acceptability of installing a system in Indian Point Unit 2. These systems serve only as an operational convenience to the plant operator and as test vehicles to evaluate the need for and suitability of incore detectors for power distribution monitoring and control. The incore detector development program was continued in the early, large plants with the principal aims of demonstrating the design lifetime of a PWR and optimizing detector parameters. Since ex-core detectors, particularly long ion chambers, have been found effective for monitoring both axial and radial gross power distribution, there were no plans to install a fixed incore system in the Surry reactors. However, provision was made so that a fixed incore detector system could be installed in the Surry reactors.

1.6.2.7 Empire States Atomic Development Associates DNB Program (Item 11 of Reference 1)

This program provided experimental rod bundle DNB data with non-uniform rod axial flux distributions. The program was conducted at Columbia University under the direction of WNES, Pittsburgh, Pennsylvania. The results of this program are detailed in WCAP-7411-L (Reference 7), which was submitted in July 1970. The experimental rod bundle data with non-uniform rod axial flux distributions are directly applicable to the design of this unit. The results of the program show that the W-3 DNB correlation applied in the Surry design is conservative.

1.6.2.8 Full Length Emergency Core Cooling Heat Transfer Test (FLECHT) (Item 12 of Reference 1)

The purpose of the FLECHT program was to investigate experimentally the thermal behavior of a simulated PWR core during the core recovery period that follows a LOCA. The first series of tests are reported in WCAP-7435 (Reference 8).

The loss-of-coolant evaluation presented in the Surry application uses conservative design assumptions in the heat transfer models for analyses of the re-flooding phase of the accident. The FLECHT program assisted in developing new analytical models to describe the core recovery phenomena. The results were favorable in 1970, at which time the program was essentially complete.

1.6.2.9 **Flashing Heat Transfer Program (Item 13 of Reference 1)**

The program is completed. It proved that the present core thermal design analysis used for evaluating the LOCA results in a conservative prediction of the peak clad temperature. The results from the program were used in the initial loss-of-coolant analysis. The program and results are summarized in WCAP-7396-L (Reference 9).

1.6.2.10 **Loss-of-Coolant Analysis Program (Item 14 of Reference 1)**

The loss-of-coolant analysis program was intended to integrate, as appropriate, the more realistic heat transfer models obtained from experimental and analytical development programs into the core thermal design codes used to evaluate the LOCA (Reference 10).

This program was completed. An evaluation of the LOCA using the results of the flashing heat transfer program in the core thermal design code is presented in WCAP-7422-L (Reference 10).

1.6.3 Assurance for Completion of Research and Development

In 1970, assurance that the necessary information would be obtained was provided by the following facts:

1. The work being done did not require development of new concepts or ideas; only the normal engineering and design work was required to complete the design.
2. Vepco and Westinghouse Electric Corporation were capable of providing necessary information in sufficient time to obtain operating licenses for the units to permit scheduled commercial operation. The research and development program was compatible with the station schedule, in that definite results would be available before the station became operational.
3. Periodic reviews of this project and other similar Westinghouse PWR projects were held with the AEC staff, as information became available, to demonstrate that the required information was being developed in a satisfactory manner.

Significant results obtained in research and development programs were formally provided to the AEC, in as timely a manner as was reasonably practicable following program completion, by the following methods:

1. Preliminary safety analysis reports on new applications.
2. Final safety analysis report on this or other applications.
3. Topical reports applicable to this and certain other applications.
4. Topical reports applicable to all applications.

1.6 REFERENCES

1. Westinghouse Electric Corporation, *Safety Related Research and Development for Westinghouse Pressurized Water Reactors - Program Summaries - Spring, 1970*, WCAP-7498-L, Westinghouse Proprietary, May 1970.
2. Westinghouse Electric Corporation, *Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Single Rod Tests*, WCAP-7379-L, Volume I, Westinghouse Proprietary, Volume II, Non-proprietary, September 1969.
3. Westinghouse Electric Corporation, *Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Multi-Rod Tests*, WCAP-7495-L, Vols. I and II, Westinghouse Proprietary.
4. Westinghouse Electric Corporation, *Loss-of-Coolant Accident: Comparison Between BLODWN-2 Code Results and Test Data*, WCAP-7401, Non-proprietary, November 1969.
5. Westinghouse Electric Corporation, *The Effects of Safety Injection on a Reactor Vessel and Its Internals Following a Loss-of-Coolant Accident*, Westinghouse Proprietary, December 1967.
6. Westinghouse Electric Corporation, *Elemental Iodine Removal by Reactive Sprays*, WCAP-7499-L, Westinghouse Proprietary.
7. Westinghouse Electric Corporation, *Rod Bundle Axial Non-Uniform Heat Flux DNB Tests and Data*, WCAP-7411-L, Westinghouse Proprietary.
8. Westinghouse Electric Corporation, *PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report* WCAP-7435, Non-proprietary, January 1970.
9. Westinghouse Electric Corporation, *Safety Related Research and Development for Westinghouse Pressurized Water Reactors - Fall, 1969*, WCAP-7396-L, Westinghouse Proprietary.
10. Westinghouse Electric Corporation, *Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident*, WCAP-7422-L, Westinghouse Proprietary.