CHAPTER 2 SUMMARY DESCRIPTION

2.1 INTRODUCTION

This Preliminary Safety Analysis Report is submitted in support of an application by Consolidated Edison Company of New York, Inc. for a permit to construct a nuclear power plant designated as Indian Point Unit No. 3. The plant is situated on the east bank of the Hudson River at Indian Point, Village of Buchanan in upper Westchester County, New York. The site is about 24 miles north of the New York City Boundary line.

The Indian Point Unit No. 3 Nuclear Generating Station will be operated by Consolidated Edison Company of New York, Inc. and will employ a pressurized water reactor nuclear steam supply system designed and furnished by Westinghouse Electric Corporation.

The reactor is designed for a power output of 3025 MWt which is the license application rating. The equivalent warranted gross and net electrical outputs of the plant are 1005 MWe and 965,300 KWe, respectively. The ultimate capability which corresponds to a valves wide open rating of the turbine generator is 1033 MWe net or an equivalent 3217 MW thermal output. All plant safety systems, including containment and engineered safeguards have been evaluated and will be designed for operation at the higher power level. The power rating of 3217 MWt is used in the accident analysis to determine containment response and off-site radiological doses in the unlikely event of a loss of coolant accident.

The design is essentially the same as Indian Point Unit No. 2 which has recently been licensed for construction by the Atomic Energy Commission for location at the same site as proposed for Indian Point Unit No. 3. All functional and safety systems for Unit No. 3 will be independent of the other units on the site, except for the common discharge canal. There are however, differences between this facility and those design features of Unit No. 2 presented in the original Indian Point Unit No. 2 Preliminary Safety Analysis Report (Docket No 50-247) and Supplements 1 through 5. In addition to power level (3025 vs. 2758 MWt), these differences are summarized as follows:

- Addition of pressurized accumulators containing a large volume of borated emergency core cooling water held back only by primary system pressure so that a drop in system pressure associated with a loss of coolant results in immediate action without dependence on any control system or power source, thus eliminating the possibility of core meltdown and the need for the reactor pit crucible.

- No reactor pit crucible will be included.

- Peak heat flux and peak linear power density have been reduced.

- Reduced hot channel factors.

- Based on further experimental work on iodine removal systems, charcoal filters have been deleted in favor of the more efficient thiosulfate spray.

- Reduced boron concentration

- Higher average fuel enrichment per core.

- Higher average burnup per core.

The changes in the reactor core design reflect additional engineering and physics studies conducted since issuance of the Provisional Construction Permit for Unit No. 2.

Refinements and additions to the Safety Injection System further reduce the probability of fuel melting in the event of a loss of coolant accident. Assessment of the Safety Injection System included in the design of Indian Point No. 2 (See Supplement VI) and proposed for inclusion in Indian Point Unit No. 3 indicate that core decay heat is adequately removed and core integrity and initial geometry are maintained. Therefore, the Indian Point Unit No. 3 preliminary design does not include a reactor pit crucible as an engineered safeguard backup.

Studies on $Na_2S_2O_3$ containment spray systems indicate that iodine absorption effectiveness is equal to or greater than removal effectiveness provided by charcoal filters. Fire harzards and handling problems associated with charcoal filters and the enhanced performance and testability of thiosulfate ($Na_2S_2O_3$) spray systems favor the use of this concept for halogen removal in Indian Point No. 3 in the unlikely event of a nuclear incident releasing fission products to the containment. Experimental verification of the analytical stuides is planned to demonstrate the ability of $Na_2S_2O_3$ solutions to remove halogens in pressurized steam-air atmospheres, determination of thermal and chemical stability, the effect of radiolysis, both on the destruction of this sulfate by gamma radiation and the possibility of thiosulfate ions altering the equilibrium hydrogen concentration in the borated water which circulates to cool the core, and the effect of thiosulfate on various materials.

The remainder of Chapter 2 of this report summarizes the principal design features and safety criteria of the nuclear unit, pointing out the similarities and differences with respect to Indian Point Unit No. 2, which has been authorized for construction at the same site and which employs the same technology and basic engineering features as the Indian Point Unit No. 3 Nuclear Station. Chapter 1, "Site and Environment", is presented under separate cover and contains a description and evaluation of the Indian Point site and environs, supporting the suitability of that site for a reactor of the size and type described. Chapters 3 and 4 describe the reactor and the reactor coolant system, Chapter 5, the containment and related systems, and Chapters 6 through 11, the emergency and other auxiliary systems. Chapter 12 is a safety evaluation summarizing the analysis which demonstrates the adequacy of the reactor protection system, and the containment and engineered safeguards systems, and which shows that the consequences of various postulated accidents are well within the guidelines set forth in the Commission's regulation 10 CFR 100.

2.2 DESIGN HIGHLIGHTS

The design of Indian Point Unit No. 3 Nuclear Station will be based upon proven concepts which have been developed and successfully applied in the construction of pressurized water reactor systems, to which have been added improved features of engineered safeguards systems designed to prevent clad melting in a major loss-of-coolant accident. In subsequent paragraphs, the design features of Indian Point Unit No. 3 Nuclear Station are indicated which represent a slight variation or extrapolations from units presently approved for construction such as Indian Point Unit No. 2 (Docket 50-247)..

2.2.1 POWER LEVEL

The license application power level for Indian Point Unit No. 3 is 3025 MWt. This increase in license application power rating over 2758 MWt for Indian Point Unit No. 2 is achieved by about 10 per cent increase in average heat flux since the reactor cores for the two units are physically identical.

2.2.2 REACTOR COOLANT LOOPS

The Reactor Coolant System for the Indian Point Unit No. 3 Nuclear Station will consist of four loops, each having components identical to those for Indian Point Unit No. 2.

2.2.3 PEAK SPECIFIC POWER

Based on the design hot channel factors, operation at 3025 MWt will yield a steady state peak specific power of 17.6 kw/ft and a corresponding peak power of 19.7 kw/ft for the 112% overpower condition. This value is justified by the results of in-core experiments by Westinghouse (Preliminary Safety Analysis Report for Indian Point Nuclear Generating Unit No. 2, Appendix A, Docket No. 50-247) and others at much higher specific power ratings. The peak ratings for Unit No. 3 are lower than the corresponding preliminary conditions for Indian Point Unit No. 2 which were 18.5 kw/ft steady state and 20.7 kw/ft overpower and are a result of lower design hot channel factors.

2.2.4 FUEL CLAD

The fuel rod design for Indian Point Unit No. 3 Nuclear Station will utilize Zircaloy as a clad material which has proved successful in the CVTR and Saxton reactors and Yankee test assemblies. The fuel rod dimensions are identical to those in Brookwood and Indian Point Unit No. 2.

2.2.5 FUEL ASSEMBLY DESIGN

The fuel assembly will incorporate the rod cluster control concept in a canless 15 x 15 fuel rod assembly utilizing a spring clip grid 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 will be available in the near future from the San Onofre and Connecticut-Yankee plants.

2.2.6 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

For a short time, less than three full power months, after the initial startup, the reactor may have a slightly positive moderator temperature coefficient of reactivity. Although this condition departs qualitatively from previous pressurized water reactor designs now operating, it is not a significant change with respect to stability, control and protection of these reactors and it matches the conditions for for Indian Point Unit No. 2. Because of the negative temperature coefficient of the fuel, the power coefficient is negative.

2.2.7 CONTAINMENT STRUCTURE

The containment structure will be a steel lined reinforced concrete cylinder and hemispherical dome roof which is supported on a base slab of reinforced concrete. The structure will be designed, fabricated

and erected in accordance with applicable codes to withstand the effects of the maximum credible earthquake, flooding conditions, windstorms, ice conditions, temperature and other deleterious natural phonomena reasonably to be anticapted at the Indian Point site during the lifetime of this unit. Access into the containment structure is limited and is provided by means of personnel air locks. An equipment hatch is also provided for major equipment removal. The design is identical to the containment for the Indian Point Unit No. 2.

2.3 SUMMARY PLANT DESCRIPTION

2.3.1 INTRODUCTION

Indian Point Unit No. 3 Nuclear Station incorporates a closed-cycle pressurized water nuclear steam supply system, a turbine-generator and their necessary auxiliaries. A radioactive waste disposal system, fuel handling system, and all auxiliaries, structures, and other on-site facilities required for a complete and operable nuclear power plant are provided. The general arrangement of the unit is shown in Site Plot Plan, Figure 2-1.

2.3.2 STRUCTURES

The major structures will be the reactor containment, primary auxiliary building, holdup tank building, turbine-generator building, and fuel handling control and admistrative facilities. Preliminary general layout of the reactor and auxiliary building and interior components arrangements is shown on Figures 2-2 and 2-3.

The reactor containment will be a steel lined reinforced concrete cylinder with a hemispherical dome and will be anchored to a reinforced concrete foundation slab. The containment will be designed to withstand the internal pressure accompanying a loss-of-coolant accident, will be virtually leak tight and will provide adequate radiation shielding for both normal operation and accident conditions.

The seismic criteria to be used to design the structures and equipment in the plant are described in Section 5. The maximum ground accelerations for the design are 0.1g horizontal and 0.05g vertical. Vertical acceleration will be considered to act simultaneously with the horizontal acceleration.

2.3.3 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system will consist of a pressurized water reactor, reactor coolant system, and associated auxiliary fluid systems. The reactor coolant system will be arranged as four closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer will be connected to one of the loops.

The reactor core will be composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The tubes will be supported in assemblies by a spring clip grid structure. The mechanical control rods will consist of clusters of stainless steel clad absorber rods and guide tubes located within the fuel assembly. The core will be loaded in three regions of different enrichments with new fuel being introduced into the outer region, moved at successive refuelings and discharged to spent fuel storage.

The steam generators will be vertical U-tube units containing Inconel tubes. Integral separating equipment will reduce the moisture content of the steam at the turbine throttle to 1/4 per cent or less.

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

Auxiliary systems will be provided to perform the following functions:

a) Charge the reactor coolant system,

b) Add makeup water,

c) Purify reactor coolant water,

d) Provide chemicals for corrosion inhibition and reactor control,

e) Cool system components,

f) Remove residual heat when the reactor is shutdown,

g) Cool the spent fuel storage pool,

h) Sample reactor coolant water,

i) Provide for safety injection,

j) Vent and drain the reactor coolant system,

k) Provide containment spray,

l) Provide containment ventilation and cooling and

m) Treat and dispose of liquid, gaseous and solid wastes.

2.3.4 REACTOR AND PLANT CONTROL

The reactor will be controlled by a coordinated combination of chemical shim and mechanical control rods. The control system will allow the plant to accept step load increases of 10% and ramp load increases of 5% per minute over the load range of 15% to, but not exceeding, 100% power under nominal operating conditions subject to xenon limitations.

Supervision and control of both the reactor and turbine generator will be accomplished from the central control room.

2.3.5 WASTE DISPOSAL SYSTEM

The waste disposal system will provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquid, gaseous and solid wastes produced as a result of reactor operation.

Liquid wastes will be collected and evaporated, and after appropriate cleaning and filtering, the evaporator condensate may be reused as reactor plant makeup water or discharged to the river via the condenser

discharge at concentrations not to exceed 10 CFR 20 limits for drinking water at the common outflow of all three units. The evaporator residues will be drummed and shipped from the site for ultimate disposal in an authorized location.

Gaseous wastes will be collected and stored until their radioactivity level is low enough so that the combined discharge to the environment from all three units will be less than 10 CFR 20 limits.

2.3.6 FUEL HANDLING SYSTEM

The reactor will be refueled by equipment designed to handle spent fuel under-water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Under-water transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat.

2.3.7 TURBINE AND AUXILIARIES

The turbine will be a tandem-compound, 4 element, 1,800 rpm unit having 44 inch exhaust blading in the low pressure elements. Six combination moisture separator-reheater units will be employed to dry and superheat the steam between the high and low pressure turbine cylinders.

Preliminary design of the auxiliaries includes three cross flow deaerating surface condensers, steam jet air ejectors, three 1/3 capacity condensate pumps, two one-half capacity boiler feed pumps and six stages of feedwater heaters.

2.3.8 ELECTRICAL SYSTEM

The main steam turbine-driven generator will be a 1,800 rpm, 3 phase, 60 cycle, hydrogen inner-cooled unit. A main step-up transformer will deliver power to the 345 KV switchyard.

The station service system will consist of auxiliary transformers, 6.9 kv. switchgear, 480 v. motor control centers, 115 v. ac instrument bus and 125 v. dc equipment.

Three diesel generator emergency sources of power (480 v), provided for the exclusive use of Unit No. 3, will be capable of operating post-accident containment cooling equipment as well as high head safety injection and residual heat removal pumps to ensure an acceptable post-loss-of-coolant containment pressure transient.

2.3.9 ENGINEERED SAFEGUARDS SYSTEMS

The engineered safeguard systems to be provided for this plant will have sufficient redundancy of components and power sources such that under the conditions of a hypothetical loss of coolant accident, the systems can, even when operating with partial effectiveness, maintain the integrity of the containment and keep the exposure to the public well within the guidelines of 10 CFR 100. With the engineered safeguards systems performing as expected, the off-site exposure resulting from the accident essentially meets the yearly exposure limits of 10 CFR 20.

The systems provided are summarized below:

a) The steel lined concrete containment vessel provides a highly reliable barrier against the escape of fission products. All containment vessel penetrations, including access openings, and ventilation ducts are provided with double containment which is continuously pressurized above containment design pressure. In addition, all containment liner welds are covered by a steel channel with the enclosed space continuously pressurized above containment design pressure. Of the pipes penetrating the containment, those which could become a potential path for leakage to the environment following a loss of coolant accident are provided with an automatically operated isolation valve and an automatic isolation valve seal water system.

b) The Safety Injection System provides borated water to cool the core by injection into the reactor coolant loops.

c) The Containment Ventilation System provides a dynamic heat sink to cool the containment atmosphere and a filtering system to remove particles, under the conditions of a loss-of-coolant accident.

d) The Containment Spray System provides a spray of cool, borated water containing a solution of sodium thiosulphate to the containment atmosphere. The spray acts as a heat sink and a means of reducing the halogen fission product concentration.

2.3.10 SERVICES AND ADMINISTRATIVE FACILITIES

Since Indian Point Unit No. 3 is located adjacent to Units No. 1 and 2 and is in essence an extension to an existing nuclear power generating facility certain common plant facilities and utilities will be extended and/or relocated for servicing the proposed plant. These include the existing parking lots, roads, potable water, fire water, sanitary sewage and the fuel oil storage facilities.

Additional adminsitrative facilities are provided for Indian Point Unit No. 3 to achieve functional separation from Units No. 1 and 2. These facilities consist of a Nuclear Service Building and a Conventional Service Building both arranged adjacent to the control building to facilitate implementation of personnel control procedures.

Approximately 6000 sq. ft. of floor space is provided in the Nuclear Service Building to house security, health physics, radio-chemistry laboratories, counting rooms, and nuclear instrument maintenance and repair facilities. The Conventional Service Building (approximately 10,000 sq. ft.) contains general administration offices, equipment storage and maintenance shops, lunch room, first aid, chemical labs and shower and change rooms.

2.3.11 DESIGN PARAMETERS

Table 2-1 presents a summary of the preliminary design and operating parameters for Indian Point Unit No. 3 Nuclear Station. The Table gives a comparison of these data with the data of the Preliminary Safety Analysis Report of the proposed Indian Point Unit No. 2 which has been approved for construction by the Atomic Energy Commission. Those items that differ significantly from Indian Point Unit No. 2 are discussed in detail following the table.

TABLE 2-1

COMPARISON OF DESIGN PARAMETERS

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
HYDRAULIC AND THERMAL DESIGN PARAMETERS			
Total Heat Output, MWt	3025	2758	1
Total Heat Output, Btu/hr	$10,324 \times 10^6$	9413×10^6	2
Heat Generated in Fuel, %	97.4	97.4	3
Maximum Overpower	12%	12%	4
System Pressure, Nominal, psia	2250	2250	5
System Pressure, Minimum Steady State, psia	2220	2220	6
Hot Channel Factors			
Heat Flux, F_q	2.82	3.25	7
Enthalpy Rise, $F_{\Delta H}$	1.70	1.88	8
DNB Ratio at Nominal Conditions	1.82 (W-3)	1.81 (W-3)	9
Minimum DNBR for Designs Transients	1.30 (W-3)	1.30 (W-3)	10
Coolant Flow			
Total Flow Rate, lb/hr	133.1×10^6	136.2×10^6	11
Effective Flow Rate for Heat Transfer, lb/hr	121.2×10^6	124.1×10^6	12
Effective Flow Area for Heat Transfer, Ft^2	47.9	48.4	13
Average Velocity Along Fuel Rods, ft/sec	15.7	16.1	14
Average Mass Velocity, lb/hr-ft^2	2.53×10^6	2.56×10^6	15
Coolant Temperatures, °F			
Nominal Inlet	549.7	543	16
Maximum Inlet Due to Instrumentation Error and Deadband, °F	553.7	547	17
Average Rise in Vessel, °F	58.0	53.0	18
Average Rise in Core	63.2	57.0	19
Average in Core	582.1	572.7	20
Average in Vessel	578.8	570.0	21
Nominal Outlet of Hot Channel	648.3	643.0	22
Average Film Coefficient, Btu/hr-ft^2-°F	5920	5900	23
Average Film Temperature Difference, °F	33	30.0	24
Heat Transfer at 100% Power			
Active Heat Transfer Surface Area, ft^2	52,200	52,200	25
Average Heat Flux, Btu/hr-ft^2	193,000	175,600	26
Maximum Heat Flux, Btu/hr-ft^2	543,000	570,800	27
Average Thermal Output, kw/ft	6.24	5.7	28
Maximum Thermal Output, kw/ft	17.6	18.5	29

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
Maximum Clad Surface Temperature at Nominal Pressure, °F	657	659	30
Fuel Central Temperature, °F			
Maximum at 100% Power	~4000	~4200	31
Maximum at Overpower	~4250	~4400	32
Thermal Output, kw/ft at Maximum Overpower	19.7	20.7	33
CORE MECHANICAL DESIGN PARAMETERS			
Fuel Assemblies			
Design	RCC Canless 15 x 15	RCC Canless 15 x 15	34
Rod Pitch, in.	0.563	0.563	35
Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426	36
Fuel Weight (as UO_2), pounds	218,530	215,319	37
Total Weight, pounds	279,631	273,408	38
Number of Grids per Assembly	9	8	39
Fuel Rods			
Number	39,372	39,372	40
Outside Diameter, in.	0.422	0.422	41
Diametral Gap, in.	0.0065	0.0065	42
Clad Thickness, in.	0.0243	0.0243	43
Clad Material	Zircaloy	Zircaloy	44
Fuel Pellets			
Material	UO_2 Sintered	UO_2 Sintered	45
Density (% of Theoretical)	94-93	94-93	46
Diameter, in.	0.3669	0.3669	47
Length, in.	0.6000	0.6000	48
Rod Cluster Control Assemblies			
Neutron Absorber	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	49
Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	50
Clad Thickness, in.	0.019	0.019	51
Number of Clusters	53	53	52
Number of Control Rods per Cluster	20	20	53
Core Structure			
Core Barrel I.D./O.D., in.	148.5/152.5	148.5/152.5	54
Thermal Shield I.D./O.D., in.	158.5/164.0	158.5/164.0	55

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
PRELIMINARY NUCLEAR DESIGN DATA			
Structural Characteristics			
Fuel Weight (as UO_2), lb.	218,530	215,319	56
Clad Weight, lbs.	41,993	43,785	57
Core Diameter, in. (Equivalent)	133.7	133.7	58
Core Height, in. (Active Fuel)	144	144	59
Reflector Thickness and Composition			
Top - Water plus Steel	10 in.	10 in.	60
Bottom - Water plus Steel	10 in.	10 in.	61
Side - Water plus Steel	15 in.	15 in.	62
H_2O/U, Unit Cell (Cold Volume Ratio)	3.48	3.48	63
Number of Fuel Assemblies	193	193	64
UO_2 Rods per Assembly	204	204	65
Performance Characteristics			
Loading Technique	3 region, non-uniform	3 region, non-uniform	66
Fuel Discharge Burnup, MWD/MTU			
Average First Cycle	13,600	12,000	67
First Core Average	22,800	21,800	68
Feed Enrichments, w/o			
Region 1	2.1	2.23	69
Region 2	2.6	2.38	70
Region 3	3.2	2.68	71
Control Characteristics			
Effective Multiplication (Beginning of Life)			
Cold, No Power, Clean	1.293	1.275	72
Hot, No Power, Clean	1.248	1.225	73
Hot, Full Power, Xe and Sm Equilibrium	1.181	1.170	74
Rod Cluster Control Assemblies			
Material	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	75
Number of RCC Assemblies	53	53	76
Number of Absorber Rods per RCC Assembly	20	20	77
Total Rod Worth	7%	7%	78

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
Boron Concentrations			
To shut reactor down with no rods inserted, Clean (k_{eff} = .99) Cold/Hot	1700 ppm/2100 ppm	3400 ppm/3500 ppm	79
To control at power with no rods inserted, Clean/Equilibrium Xenon and Samarium	1800 ppm/1500 ppm	2800 ppm/2300 ppm	80
Boron Worth, Hot	1% $\delta k/k$ / 85 ppm	1% $\delta k/k$ / 150 ppm	81
Boron Worth, Cold	1% $\delta k/k$ / 70 ppm	1% $\delta k/k$ / 120 ppm	82
Kinetic Characteristics			
Moderator Temperature Coefficient	$+1.0 \times 10^{-4}$ to -3.0×10^{-4} $\delta k/k$ / °F	$+1.0 \times 10^{-4}$ to -3.0×10^{-4} $\delta k/k$ / °F	83
Moderator Pressure Coefficient	-1.0×10^{-6} to $+3.0 \times 10^{-6}$ $\delta k/k$ / psi	-1.0×10^{-6} to $+3.0 \times 10^{-6}$ $\delta k/k$ / psi	84
Moderator Void Coefficient	$+1.0 \times 10^{-3}$ to -3×10^{-3} $\delta k/k$ / % void	$+1.0 \times 10^{-3}$ to -3×10^{-3} $\delta k/k$ / % void	85
Doppler Coefficient	-1×10^{-5} to -2.0×10^{-5} $\delta k/k$ / °F	-1×10^{-5} to -2.0×10^{-5} $\delta k/k$ / °F	86
REACTOR COOLANT SYSTEM - CODE REQUIREMENTS			
Component			
Reactor Vessel	ASME III Class A	ASME III Class A	87
Steam Generator			
Tube Side	ASME III Class A	ASME III Class A	88
Shell Side	ASME III Class C	ASME III Class C	89
Pressurizer	ASME III Class A	ASME III Class A	90
Pressurizer Relief Tank	ASME III Class C	ASME III Class C	91
Pressurizer Safety Valves	ASME III	ASME III	92
Reactor Coolant Piping	ASA B31.1	ASA B31.1	93

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM			
Reactor Heat Output, MWt	3025	2758	94
Reactor Heat Output, Btu/hr	$10,324 \times 10^6$	9412×10^6	95
Operating Pressure, psig	2235	2235	96
Reactor Inlet Temperature	549.7	543	97
Reactor Outlet Temperature	607.5	596	98
Number of Loops	4	4	99
Design Pressure, psig	2485	2485	100
Design Temperature, °F	650	650	101
Hydrostatic Test Pressure (Cold), psig	3110	3110	102
Coolant Volume, including pressurizer, cu.ft.	12,209	12,209	103
Total Reactor Flow, gpm	354,000	358,800	104
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL			
Material	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	105
Design pressure, psig	2485	2485	106
Design Temperature, °F	650	650	107
Operating Pressure, psig	2235	2235	108
Inside Diameter of Shell, in.	173	173	109
Outside Diameter Across Nozzles, in.	254	245	110
Overall Height of Vessel & Enclosure Heat, ft-in.	42-4	42-4	111
Minimum Clad Thickness, in.	5/32	5/32	112
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS			
Number of Units	4	4	113
Type	Vertical U-Tube with integral-moisture separator	Vertical U-Tube with integral-moisture separator	114
Tube Material	Inconel	Inconel	115
Shell Material	Carbon Steel	Carbon Steel	116
Tube Side Design Pressure, psig	2485	2485	117
Tube Side Design Temperature, °F	650	650	118
Tube Side Design Flow, lb/hr	33.28×10^6	34.05×10^6	119

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
Shell Side Design Pressure, psig	1085	1085	120
Shell Side Design Temperature, °F	600	600	121
Operating Pressure, Tube Side, Nominal, psig	2235	2235	122
Operating Pressure, Shell Side, Maximum, psig	1005	1005	123
Maximum Moisture at Outlet at Full Load, %	1/4	1/4	124
Hydrostatic Test Pressure, Tube Side (cold), psig	3110	3110	125
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS			
Number of Units	4	4	126
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	127
Design Pressure, psig	2485	2485	128
Design Temperature, °F	650	650	129
Operating Pressure, Nominal, psig	2235	2235	130
Suction Temperature, °F	545	543	131
Design Capacity, gpm	88,500	89,700	132
Design Head, ft.	277	272	133
Hydrostatic Test Pressure (cold), psig	3110	3110	134
Motor Type	A-C Induction single speed	A-C Induction single speed	135
Motor Rating	6000 HP	6000 HP	136
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING			
Material	Austenitic SS	Austenitic SS	137
Hot Leg - I.D., in.	29	29	138
Cold Leg - I.D., in.	27-1/2	27-1/2	139
Between Pump and Steam Generator - I.D., in.	31	31	140
Design Pressure	2485	2485	141

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
CONTAINMENT SYSTEM PARAMETERS			
Type	Steel lined, reinforced concrete, vertical cylinder with flat bottom and hemispherical dome	Steel lined, reinforced concrete, vertical cylinder with flat bottom and hemispherical dome	142
Design Parameters			
Inside Dia., ft.	135	135	143
Height, ft.	212	212	144
Free Volume, ft^3	2,610,000	2,610,000	145
Reference Incident Pressure, psig	47	47	146
Reference Incident Energy, BTU (E_1)	305,290,000	305,290,000	147
Energy Required to Produce Incident Pressure (E_2)	349,880,000	349,880,000	148
Ratio: E_1/E_2	0.873	0.873	149
Ratio: $(E_2 - E_1)E_1$	0.146	0.146	150
Concrete Thickness ft.			
Vertical Wall	4-1/2	∿ 5-1/2	151
Dome	3-1/2	∿ 4-1/2	152
Containment Leak Prevention and Mitigation Systems	Continuously pressurized double penetrations, liner weld channels and access openings; isolation valve seal water system automatically isolates piping, where required. Continuous leak rate monitoring of containment and pressurized areas. None of above require outside power to operate.	Continuously pressurized double penetrations, liner weld channels and access openings; isolation valve seal water system automatically isolates piping, where required. Continuous leak rate monitoring of containment and pressurized areas. None of above require outside power to operate.	153
Gaseous Effluent Purge	Vent discharge from top of containment (∿ 150' above grade)	Vent discharge from top of containment (∿ 150' above grade)	154
ENGINEERED SAFEGUARDS			
Safety Injection System			
No. of Accumulators	4	4 (Proposed in Docket No. 50-247, Supplement 6)	155
No. of high head pumps	3	3	156
No. of low head pumps	2	2	157
Containment Fan Coolers			
No. of units	5	5	158
Air flow capacity, each, at accident condition	65,000 cfm	65,000 cfm	159
Post-Accident Filters			
No. of units	5	5	160
Air flow capacity, each, at accident condition	65,000 cfm	65,000 cfm	161
Type	Roughing/absolute	Roughing/absolute/charcoal	162

TABLE 2-1 (Continued)

	INDIAN POINT #3 PRELIMINARY REPORT	INDIAN POINT #2 PRELIMINARY REPORT	REFERENCE LINE NO.
Filtration reduction rate $\eta_f R/V_c$ (η_f - 0.9 per pass)	---	6.75 hr^{-1} (5 units)	163
Containment Spray			
No. of Pumps	2 (See Note 1)	2 (See Note 1)	164
Emergency Power			
Diesel-Generator Units	3	3	165
Engineered Safeguards Operable	(2 of 3 diesels)	(2 of 3 diesels)	166
From Diesels (Minimum)	2 high head SI pump	1 high head SI pump	167
	1 low head SI pump	1 low head SI pump	168
	3 fan-cooler units	4 fan-cooler units	169
	1 spray pump	1 spray pump	170
	1 service water pump	1 service water pump	171

Note: Spray contains chemical reagent to halogen removal

Comparison of Design Parameters

I Thermal and Hydraulic Design Parameters:

The only items which have changed significantly from the Indian Point Unit #2 design as presented in that application are the power level (Line item #1), hot channel factors (Line Items #7&8) and heat flux and average linear power density (Line Item #26-29). These differences are inter-related and result from improvements in the hot channel factors for the Indian Point #3 core. Other minor changes are due to more advanced design information.

The power level is about 10% higher (3025 MWt vs 2758 MWt) and this results in a similar increase in average heat flux (193,000 Btu/hr ft^2). The heat flux hot channel factor has been reduced from 3.25 to 2.82 and hence the peak heat flux has been reduced. This reduction in peak heat flux permits operation with higher core inlet temperature and higher average heat flux while maintaining essentially the same margin to DNB.

The reduction in the overall hot channel factor is due to a reduction in the nuclear hot channel factor. The nuclear hot channel factors employed in previous PWR core designs were based upon preliminary values established at the initiation of the San Onofre development program in 1960 (A.E.C. LRD Contract AT (30-1) -3269). Since that time, the LRD program has been completed and substantial fuel management analyses have been performed on many cores, including Indian Point Unit #2.

In addition, experience has been gained in the operation of the SELNI reactor. The measured hot channel factors in this reactor at beginning of life were F_q = 2.16 and $F_{\Delta H}$ with all rods out and F_q = 2.58 and $F_{\Delta H}$ = 1.60 with the control bank halfway inserted. These measurements were within 2% of the calculated values.

All of the information developed in the past six years leads to the conclusion that the indicated reduction in hot channel factors can be achieved. A substantial part of this reduction can be attributed to the development of the RCC fuel assembly design with the attendent reduction in local peaking.

It should be noted that the hot channel factors will be verified with the in-core instrumentation during initial operation of the plant and before the initial rise to full power.

The effective flow area (Line Item 13) is reduced approximately 1% due to a revaluation of the flow distribution in the core. The geometric flow area is unchanged.

II Core Mechanical Design Parameters

The mechanical design parameters are the same as current Indian Point Unit #2 values. A more precise evaluation of the fuel assembly weight since issuance of the original Indian Point Unit #2 Safety Analysis Report (Docket No. 50-247) has changed the fuel, clad, and total core weights; (218,530 vs 215,319 #UO_2), (41,933 vs 43,785) and (279,631 vs 273,408 #). Also, the number of fuel assembly support grids has been increased from 8 to 9 per assembly.

III Nuclear Design Parameters

The changes in the nuclear design parameters are in the enrichments (Line Items #69, 70 & 71), effective multiplication (Line Items #72, 73 & 74), and the Boron concentration requirements (Line Items #79, 80, 81 & 82). The nuclear design parameters for Indian Point Unit No. 3 reflect current core physics information. Those for Unit No. 2 were based on preliminary physics information.

At the time of the Indian Point Unit No. 2 Preliminary Safety Analysis Report Submittal, the work on the characteristics of zirconium clad cores had not progressed sufficiently to provide precise core physics data in all areas. Work on values related to safety was emphasized. Since the moderator temperature coefficient and its potential reactivity insertion is related to the amount of reactivity controlled by boric acid in the moderator, rather than the concentration of boric acid itself, boric acid concentrations were not updated as work progressed. The values retained were conservative in terms of the requirements and performance of the Chemical and Volume Control System. The boron concentration numbers originally quoted on the basis of old design calculations were arbitrarily increased over the best estimate values to be conservative, whereas the Unit No. 3 numbers represent the best estimate values.

The neutron multiplication values have been revised according to recent studies of zirconium core characteristics. These values represent increases which suggest, of course, slightly more control in boric acid in the moderator and therefore more positive reactivity effects. Sufficient conservatism remains in the quoted values of moderator reactivity effects in the Indian Point Unit No. 2 PSAR to account for this revision. These values, therefore, have not been adjusted for the Indian Point Unit No. 3 core.

IV Reactor Coolant System Conditions and Components

There are no significant differences in Indian Point 2 and 3 plants regarding the reactor coolant system conditions (pressure and temperature) or the design of the reactor coolant system components (reactor vessel, pumps, pressurizer, steam generators and piping).

2.4 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA FOR DESIGN

The following principal design criteria will be followed in the design of Indian Point No. 3 Nuclear Station. They have been developed as performance criteria which define or describe safety objectives and approaches rather than enumerate specific safety devices and procedures, and they provide a guide to the type of plant design information which is included in this report. Each criterion is followed by a brief summary of the preliminary design and procedures which are described more fully in other Chapters of this report and which are intended to meet the design objectives reflected in the design criterion. Some refinements and modifications in the preliminary design and procedures, consistent with the design criteria are to be expected in the course of detailed plant design. The parenthetical number following the criterion title indicates the number of its related proposed AEC criterion as published in its General Design Criteria for Nuclear Power Plant Construction Permits.

2.4.1 QUALITY AND PERFORMANCE STANDARDS (1)

Those features of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences shall be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. Approved design codes shall be used when appropriate to the nuclear application.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, or other natural phenomena characteristic of the proposed site.

Features of the facility essential to accident prevention and mitigation are 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 which depressurize and reduce the contamination level of the containment; power supplies and essential services to the above features; and the components employed to safely convey and store radioactive wastes and spent reactor fuel.

Quality standards of material selection, design, fabrication, and inspection governing the above features will conform to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the reactor containment will conform to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is given in Chapter 5.0. Vessels will comply with Section III of the ASME Boiler and Pressure Vessel Code under the specific classification dictated by their use. The principles of this Code, or equivalent guidelines, will be 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 will conform to the requirements of USA Standards Institute (USA S.I.) standard B 31.1.

Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material 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 which is more than four times the probable number of such cycles, is less than that at which propagation of material defects would occur. Design margin and material surveillance ensure that the vessel will be operated well within the ductile range of temperatures when vessel stresses are above 10,000 psi. The reactor vessel size is within the range of previous experience of the manufacturer and of the nuclear plant designer. Further discussion of quality assurance in the reactor vessel is given in Chapter 4.

Consolidated Edison Company of New York, Inc. or its agents will review specifications and perform inspections during fabrication to assure a high degree of quality.

All piping, components and supporting structures of the reactor and safety related systems are designed to withstand any seismic disturbance predictable for the site. These criteria specify that there will be no loss of function of such equipment in the event of a hypothetical ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of the structure to ground acceleration, based on appropriate spectral characteristics of the site foundation soils and on the damping of the foundation and structure, is included in the design analysis.

The reactor containment is defined as a Class I structure. Its structural members will have sufficient capacity to accept without exceeding yield stresses a combination of normal operating loads, functional loads due to a loss of coolant accident, and the loadings imposed by the maximum wind velocity, or those due to design earthquake, whichever is the larger.

The emergency on-site power sources are not subject to interruption due to earthquake, windstorm, floods, or to disturbances on the external power grid. Power cabling, motors and other equipment required for operation of the engineered safeguards are suitably protected against the effects of the accident, or of severe external weather conditions as applicable, to obtain a high degree of confidence in the operability of these systems in the event they should be required.

2.4.2 RELIABILITY (15 & 16)

Sufficient redundancy and independence shall be provided in systems so that no single failure of any active component of the system can prevent action necessary to prevent an unsafe condition.** These systems should be designed so that effects of such conditions as gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (extreme heat, cold, fire, steam, water, etc.) cause the system to go into its safest state (fail-safe) or are tolerable on some other basis. Redundancy and independence of static elements such as piping and wiring are necessary only if the event to be protected against can cause damage to the static element and thereby prevent a necessary safety action.

The reactor protection system will be designed using the proposed IEEE "Standards for Nuclear Power Plant Protective Systems" as a guide. Every protection channel is at least duplicated (a one-out-of-two trip mode) and, in all cases for protection during power operation, at least triplicated (a two-out-of-three trip mode). The startup rate trip channel is one-out-of-two, the engineered safeguards trip channel is two-out-of-three and the nuclear overpower trip channel is two-out-of-four. The coincident trip philosophy (two of three or two of four) is carried out to provide a safe and reliable system since a single failure will not defeat the function of the channel and will also not cause a spurious plant trip. Channel independence is carried throughout the system extending from the sensor to the relay providing the two-out-of-three logic. The power supplies to the channels are fed from instrumentation vital buses. The vital buses are automatically transferred to voltage regulated motor control center sources if the normal inverter source fails.

Two reactor trip breakers are provided to interrupt power to the rod drive mechanisms. The breaker main contacts are connected in series with one contact from each breaker on either side of the mechanism coils.

** As used in these criteria, an unsafe condition means a condition which would increase significantly the likelihood of release of unacceptable quantities of radioactivity to the public environment. The term also takes into account any significant increase in the likelihood of exposing the public to unacceptable levels of direct radiation. Unacceptable quantities of radioacitivty release and unacceptable levels of radiation exposure under both normal and abnormal circumstances have been defined by the AEC in 10 CFR 20 and 10 CFR 100, respectively.

coils. Opening either breaker will interrupt DC power to all mechanisms causing them to release all rods to fall by gravity into the core. Each breaker is actuated with a shunt trip coil and an undervoltage trip coil. Each protection channel feeds three two-out-of-three relay matrices, one for each shunt trip coil and one for the undervoltage trip circuits.

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 would cause the system to fail safe and go into its safety state. Reliability and independence is obtained by redundancy within each channel. In a two-out-of-three circuit, for example, the three channels are equipped with separate transmitters and energized by separate electrical circuits. Failure to de-energize when required would be a mode of malfunction likely to affect only one channel; the trip signal furnished by the two remaining channels would be unimpaired in this event.

Control rod cluster insertion is itself a fail-safe function. Reactor trip is implemented by interrupting DC power to the magnetic latch mechanisms on each drive, allowing the rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of vital DC power.

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

The initiation of the engineered safeguards systems provided for loss-of-coolant accidents, e.g., high head safety injection and residual heat removal pumps, containment air recirculation coolers, and containment spray systems, is accomplished from redundant signals derived from reactor coolant system and containment instrumentation. Each of three pressurizer pressure instruments and each of three pressurizer water

level instruments sends a signal to a relay matrix which develops an engineered safeguards trip signal when two-out-of-three low pressure signals are received in coincidence with two-out-of-three low water level signals. Channel independence is carried throughout the system from the sensors to the signal output relays including the power supplies for the channels. The safety injection accumulators normally function on opening of a self actuated check valve, hence requiring no actuating signal. The initiation signal for containment spray comes from coincident high containment pressure and safety injection signals. The containment air recirculation coolers are normally operating and therefore do not require any initiation signal. Signals from safety injection on high containment pressure will actuate the air recirculation coolers in the event they are not operating.

The signal for containment isolation, i.e., the isolation valves trip signal, is derived from a coincidence of two-out-of-three containment high pressure signals. For this circuit also the channels are independent from sensor to output relay and are supplied from at least two independent power sources. Failure of any channel component will cause that channel to fail safe and go into its safety state.

2.4.3 TESTING (15, 20 & 23)

Capability shall be provided for demonstrating by analysis or test the functional operability of systems or components necessary to prevent an unsafe condition.

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

2.4.3.1 Initial Performance Tests

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

For example, pumps will be tested in the manufacturer's shops to establish conformance with design conditions. Similarly, system valves will be tested in the shop to verify opening and closing times and the ability of the valve operator to initiate motion with the maximum anticipated differential hydrostatic pressure applied where applicable. Cooling units and demisters of the type and design to be used will also be tested to demonstrate conformance to design conditions.

The initial tests of the Containment Spray System and the Safety Injection System will be conducted prior to power operation. This test will complement the shop tests of individual components. No attempt will be made to achieve flows approaching the maximum values which were demonstrated for pumps in the shop tests. The purpose of the integrated system test will be to demonstrate proper functioning of instrumentation and actuation circuits, to evaluate the dynamics of placing the system in operation, and to expose all members in the system to pressure conditions representative of those which can be expected for a loss-of-coolant accident.

Flow will not be introduced into the containment spray nozzles during this test but will be established in all parts of the system up to the final remote operated isolation valves in the containment spray loop. Permanent test piping is installed between a point upstream of the final isolation valve (this valve must be unlocked and manually closed) to the minimum flow recirculation line to verify flow. The remote operated valves in the containment spray system will be tested separately. Nozzle clearance will be verified by introducing air or smoke in the spray header.

Safety Injection System operation will be initiated by the installed instrumentation and controls. Both pressurizer level and pressure will be varied to provide the required coincidence of low level and pressure to initiate injection, or set points of pressure and level bistable units will be varied to produce an automatic safety injection signal. Containment spray operation will be similarly tested with the addition of a simulated containment high pressure signal.

Valve operating times, system flows, and system pressures will be measured. In addition, pump acceleration times and associated auxiliary electrical system voltage dips will be measured. Upon actuation of the safety injection signal, complete flow paths will be aligned, and flow will be introduced into the Reactor Coolant System. Rising water level in the pressurizer will verify flow into the system.

After installation at the site, the emergency power sources will be tested for conformance to design requirements. In addition, tests will be performed to assure that the desired automatic sequence of connecting the emergency power sources to the buses, and placing the required engineered safeguards into operation will be successfully accomplished.

2.4.3.2 Periodic Testing

The following series of periodic tests and checks can be conducted to assure that the systems can perform their design functions whenever they should be called on during the plant lifetime.

a) Integrated Test Actuation Circuits and Motor-Operated Valves

The automatic actuation circuitry, valves and pump breakers can be checked during integrated system tests performed during each planned cooldown of the Reactor Coolant System for refueling.

The integrated system test can be performed during the late stages of plant cooldown when the residual heat removal loop is in service. This test would not introduce flow into the Reactor Coolant System but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

b) Accumulator Tanks

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

c) Safety Injection, Residual Heat Removal, Containment Spray and Charging Pumps Tests

The safety injection and containment spray pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loops are put into operation. The charging pumps are normally run during plant operation. All remote operated valves can be exercised and actuation circuits can be tested periodically during plant operation or routine maintenance.

d) Air Recirculation and Cooling Units

The air recirculation and cooling units and the service water pumps that supply the cooling units are in operation on a relatively continuous schedule during plant operation, and no additional periodic test is required. In place bypass efficiency tests of the absolute filter units will be conducted periodically using a test aerosol.

e) Boric Acid Concentration in the Accumulators and Injection Lines

The accumulators and the safety injection piping up to the final isolation valve is maintained full of borated water at refueling water concentration while the plant is in operation. This concentration

will be checked periodically by sampling. The accumulators and injection lines will be refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small bypass line and a return line are provided for this purpose in each injection flow path.

f) Sodium Thiosulfate Concentration in the Thiosulfate Tank

The concentration of $Na_2S_2O_3$ solution in this tank will be checked periodically by local sampling. Additional solution can be added through a connection provided for this purpose.

g) Emergency Power Sources

The emergency power generators can be tested at any time by starting the units from local control panels. The test will be initiated by simulating a loss of voltage on the undervoltage relays. Automatic starting of the emergency generators and delivery of the required out put voltage will be verified as a function of time. Also, load carrying capability and proper sequencing of critical loads will be verified periodically.

h) Containment Penetrations and Weld Channels

All penetrations are designed with double seals so as to permit continuous pressurization of the interior of the penetration. All containment welds are backed by a steel channel to provide a space for continuous pressurization. The large access openings such as the equipment hatch and personnel air locks are equipped with double gasket seals with the space between the gaskets to permit continuous pressurization between the gaskets. The containment penetrations and weld channels are provided with pressure and makeup flow instrumentation which continuously monitor the effectiveness of these features.

i) Instrumented Protection Channels

All reactor protection channel circuits are supplied in multiples which provide the capability for channel calibration and test at any time. Bypass removal of one trip circuit places that circuit in a half-tripped mode; i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. In every case but the startup rate trip, testing will not trip the system unless a trip condition exists in a concurrent channel.

Each protection channel circuit in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation. Testing can be accomplished of individual system components up to the final relay which forms the two-out-of-three logic. Thus, the operability of each trip channel can be determined conveniently and without ambiguity. During each refueling, a complete channel test will be performed through and including the final trip breakers.

Initiation of the engineered safeguards systems and containment isolation also employs coincidence circuits which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel places that circuit in the half-tripped mode.

2.4.4 CONTROL (13 & 14)

The reactor facility shall be designed so that all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times.

The Indian Point Unit #3 Nuclear Station will be equipped with a control room which will contain all controls and instrumentation necessary for operation of the reactor and turbine generator under normal or accident conditions.

Shielding will be provided, as required, to insure that an operator in the control room will not receive more than 2.5 rem integrated whole body dose during the four weeks following an accident. The control room will also be designed to limit the thyroid exposure of any operator to less than 300 rem during the course of the accident. The above dose rates allows access to and egress from the control room and for inspection of plant equipment for such a time that the maximum potential whole body dose will not exceed 25 rem.

2.4.5 ELECTRIC POWER SUPPLIES (21)

Sufficient normal and emergency supplies of electrical power shall be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

Indian Point Unit #3 Nuclear Station will be supplied with normal, standby, and emergency power with four separate and independent sources available as follows:

a) Normal source of auxiliary power during plant operation is the main generator. Power will be supplied by a unit auxiliary power transformer that is connected to the main leads of the generator.

b) Stand-by power required during plant startup, shutdown, and after reactor trip will be supplied from a 138 kv/6.9 kv station auxiliary transformer connected to the existing 138 kv Buchanan Substation bus.

c) Emergency power will be automatically available from three emergency diesel generators.

d) The emergency supply for instruments and control will be provided by 125 V, dc, station batteries.

2.4.6 PROTECTION AGAINST DYNAMIC EFFECTS (3)

Protection shall be provided against dynamic effects resulting from plant equipment failures and causing an unsafe condition.

The high pressure equipment in the reactor coolant system will be protected by barriers to prevent a missile, generated from the reactor coolant system in a loss-of-coolant accident, from reaching either the containment liner or the containment cooling equipment, and from impairing the function of the engineered safeguards systems.

The principal missile barriers will be the concrete operating floor and the reinforced concrete shield wall enclosing the reactor coolant loops. The steam generator secondary shell will provide additional protection from missiles originating in the reactor compartment.

The pressurizer will be protected by a steel and concrete missile barrier constructed above the operating floor. A steel and concrete structure will also be provided over the control rod drive mechanisms to block a missile generated from fracture of the mechanisms.

In the evaluation of the detailed layout of the high pressure equipment and barriers which afford missile protection, fluid and mechanical driving forces will be calculated, and consideration will be given to the possibility of damage due to water jets and possible secondary missiles.

Protection from the following types of missiles will be provided: Valve stems and bonnets, instrument thimbles, nuts and bolts, and complete control rod drives.

2.4.7 NIL-DUCTILITY TEMPERATURE LIMITS (11)

Components of the primary coolant and containment systems which are potentially subject to propagation-type failure shall be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless their temperatures are sufficiently above the nil-ductility temperatures.

The design transition temperature (DTT) for the reactor vessel material before irradiation will be specified after the actual nil ductility properties of the materials have been determined. The specified DTT will be a minimum of NDT temperatures plus 60°F and will dictate the procedures followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of DTT will be increased during the life of the plant as required by the expected shift in the NDT temperature, confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime.

For reactor vessel design purposes, a DTT shift of 275°F has been selected as a value which will allow plant operation without undue limitations and provide an adequate margin between the maximum DTT and the normal reactor operating temperature. The DTT shift of 275°F corresponds to an integrated fast flux ($E > 1$ Mev) of approximately 3.7×10^{19} n/cm^2 which is in excess of the expected exposure of this vessel of 1.4×10^{19} n/cm^2 throughout the plant lifetime. During the detailed design of the plant, the integrated fast flux will be calculated to verify that it is below 3.7×10^{19} n/cm^2.

To define permissible operating conditions below DTT, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. As the normal operating temperature of the reactor vessels will be well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure. Chapter 4 contains further discussion of nil-ductility consideration in the reactor vessel design.

The concrete vapor container is not susceptible to a low temperature brittle fracture.

The containment liner is enclosed within the containment and thus will not be exposed to the temperature extremes of the environs. The containment ambient temperature during operation will be between 50 and 120°F which is expected to be well above the NDT temperature + 30°F for the liner material. Containment penetrations which can be exposed to the environment will also be designed to the NDT + 30 Criterion.

2.4.8 REACTOR PROTECTION SYSTEM (15)

A reliable protection system shall be provided to automatically initiate appropriate action whenever such action is necessary to prevent an unsafe condition.

During reactor operation in the startup and power modes redundant safety limit signals will automatically actuate two reactor trip breakers which are in series with the rod drive mechanism coils. This action will interrupt power and initiate reactor trip. This criterion as applied to the Reactor Protection System is discussed more fully in Section 2.4.2.

Automatic starting of the emergency generators will be initiated by undervoltage relays to which the emergency power sources will be connected. Engine cranking will be accomplished by a stored energy system supplied solely for the associated emergency generator.

2.4.9 OSCILLATIONS AND TRANSIENTS (4 & 5)

The reactor system shall be designed to accommodate or readily suppress, without causing an unsafe condition, oscillations or transients resulting from anticipated events such as tripping of the turbine generator or loss of power to the reactor recirculation pumps.

The reactor protection system will be designed to actuate a reactor trip for any credible combination of plant conditions which can cause the Departure from Nucleate Boiling Ratio (DNBR) to be less than 1.30 and possible fuel failure or coolant system damage.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer will result in a high pressurizer pressure trip and thereby prevent fuel damage for this transient. A loss of load of 40% of full power or less will normally be controlled by rod cluster insertion together with a controlled steam dump to the condenser to prevent a large temperature and pressure increase in the reactor coolant system and thus prevent a reactor trip. In this case, the overpower-overtemperature protection will avoid any combination of pressure, temperature and power which could result in a DNB ratio less than 1.3 during the transient.

The reactor coolant pumps to be provided for the Indian Point Unit #3 Nuclear Station will have sufficient rotational inertia to maintain an adequate flow coastdown in the event of simultaneous loss of power to all pumps. The amount of required inertia is established when detailed system design parameters are available. The flow coastdown inertia will be sufficient so that the reduction in heat flux obtained with a low flow reactor trip will prevent core damage.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and small void fraction. Protection circuits actuated directly by the coolant conditions identified with core limits will therefore be effective in preventing core damage.

2.4.10 FUEL PERFORMANCE (6)

The fuel shall be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without failure that would result in fission product inventories in the primary coolant or in storage facilities that would preclude continued operation within the limits imposed by applicable regulations for normal release and potential accident releases.

The integrity of the fuel cladding will be ensured by preventing excessive fuel swelling, excessive clad overheating, and excessive cladding stress. This will be achieved by designing the fuel elements so that the following conservative limits will not be exceeded during anticipated normal and abnormal operating conditions (including design overpower of 112%):

a) Minimum DNB ratio equal to or greater than 1.3
b) Fuel center temperature below melting point of UO_2
c) Clad stresses less than the Zircaloy yield strength

The fuel rod will be designed such that the internal gas pressure will be less than the nominal external pressure (2250 psia), even at the end of life.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions will be shown by analyses described in Chapter 12 to satisfy the demands of plant operation well within applicable regulatory limits.

2.4.11 REACTIVITY INSERTION (7)

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted shall be held to values such that no single credible control system malfunction could cause a reactivity transient capable of causing an unsafe condition.

The reactor control system will employ 53 control rod clusters, slightly less than half of which are fully withdrawn during power operation, serving as shutdown rods. The remaining rods will comprise the controlling groups, and will be used to control load and reactor coolant temperature. The rod cluster drive mechanisms will be wired into preselected groups, and will therefore be prevented from being withdrawn in other than their respective groups. The rod drive mechanism will be of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel with a mechanical limit on the maximum rod withdrawal speed. The maximum reactivity insertion rate will be analyzed in the detailed plant analysis assuming the highest worth group to be accidentally withdrawn at its maximum speed, yielding reactivity insertion rates of the order of $2 \times 10^{-4} \Delta k/sec$ which is 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 rod cluster to be withdrawn at a speed greater than its mechanical limit. The failure of a control rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not as 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 would be less severe than from the hypothetical loss of coolant, for which public health and safety are shown to be adequately protected.

2.4.12 CONTROL ROD EJECTION (-)

The reactor shall be designed and operated so that a control rod ejection brought about by failure of a rod drive housing does not cause further rupture of the primary system.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution,

only the rod cluster control assemblies in the controlling group are inserted in the core at full power, and these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to insure that this condition is met. By utilizing the available flexibility in the selection of control rod groupings, radial locations and position as a function of load, the final design will limit the maximum fuel temperature for the highest worth ejected rod to a value which will preclude any consequential damage to the primary system, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges. Limited fuel damage would occur for this accident.

2.4.13 SHUTDOWN MARGIN (8)

Reactivity shutdown capability shall be provided to make the core subcritical from any credible operating condition with the most reactive control rod withdrawn.

The maximum excess reactivity expected for the Indian Point Unit #3 Nuclear Station core is 0.293 and occurs for the cold, clean condition at the beginning of life of the initial core. This excess reactivity will be controlled by a combination of control rods and soluble neutron absorber (boron). A total of 53 Rod Cluster Control (RCC) assemblies are provided. These assemblies are divided into two categories comprising a control group and shutdown groups.

The control group, used in combination with chemical shim control provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies will be used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup.

The shutdown groups are provided to supplement the control group of RCC assemblies to make the reactor at least one per cent subcritical (k_{eff} = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position. Manually controlled boric acid addition will be used to supplement the RCC assemblies in maintaining the shutdown margin for the long term conditions of xenon decay or plant cooldown. See Section 9.1 concerning details of the Chemical and Volume Control System.

2.4.14 PRIMARY SHUTDOWN SYSTEM CAPABILITY (12)

The primary shutdown system shall be designed to be operable under abnormal conditions anticipated at the site.

The Indian Point Unit #3 Nuclear Station reactor will use the Westinghouse magnetic latch-type control rod drive mechanisms which are the same type as those used in the San Onofre, Ginna, Indian Point #2 and Connecticut Yankee plants. Upon a loss of power to the coils, the rod cluster control assembly will be released and fall by gravity into the core.

The reactor internals, fuel assemblies, RCC assemblies and drive system components will be considered Class I for seismic design purposes as defined in Appendix A. The RCC assemblies will be fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Due to this and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

The RCC control rod guide system throughout its length is locked together with dowels to insure against misalignments which might impair control rod movement under normal operating conditions and credible accident

conditions. An analogous system has successfully undergone 4132 hours of testing in the Westinghouse Reactor Evaluation Channel during which about 27,200 feet of step-driven travel and 1461 trips were accomplished with test misalignments in excess of the maximum possible misalignment that may be experienced when installed in the plant.

2.4.15 SECONDARY SHUTDOWN CAPABILITY (9)

Secondary or backup reactivity shutdown capability shall be provided that is independent of primary means of reactivity shutdown. This system must have the capability to shutdown the reactor from any operating condition.

Normal reactivity shutdown capability will be provided by control rods with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. Any time that the plant is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection will always exceed that quantity required for the normal cold shutdown. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

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

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

2.4.16 DECAY HEAT DISSIPATION (10)

The design shall provide means of dissipating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser or complete or partial loss of primary coolant from the reactor.

Redundant heat removal systems will be provided that are capable of removing core decay heat following all anticipated abnormal and credible accident conditions.

The sequence of events following loss of all auxiliary a.c. power is as follows: reactor trip and turbine trip would occur resulting in opening of the steam bypass valve to the main condenser. The subsequent loss of condenser vacuum signal would then close the bypass valve. Continued pressure relief would be accomplished by discharging the non-radioactive steam through the atmospheric dump valves and for a short time through the safety valve. Water supplied by either the turbine-driven auxiliary feedwater pump or the auxiliary feedwater pumps driven by emergency power, would be sufficient to restore the normal water level in the steam generators without uncovering the steam generator tube sheet and to maintain the plant in the hot shutdown condition. Heat would be transferred from the core to the steam generators by natural circulation of the reactor coolant. The supply of stored condensate will be adequate to dissiplate decay heat in this manner for at least 24 hours.

Following a loss-of-coolant accident, heat removal from the reactor core and from the containment atmosphere would be accomplished by the engineered safeguards systems, described in Chapter 6.

The Safety Injection System will initially supplement the reactor coolant inventory with water from the accumulator tanks and the refueling water storage tank. Heat is transferred to the water in the core, resulting in the formation of some steam and raising the temperature of the injected water, some of which subsequently spills to the containment sump. When a predetermined fraction of the water in the refueling water storage tank is depleted, water from the sump is recirculated to the reactor coolant system through the residual heat exchangers; the component cooling loop of the auxiliary coolant system is the means for transferring residual heat from these heat exchangers to the service cooling water system and thence to the river. The emergency generators can power the necessary equipment to accommodate the removal of heat (both short term and long term) from the reactor core and containment under loss-of-coolant accident conditions.

Some of the core residual heat will cause steam to be released to the containment atmosphere by injection water evaporation so that all of the residual heat will not be removed from the containment by the recirculation of the water in the sump. This heat released to the containment atmosphere will be removed by the fan cooler units.

Two means of removing heat from the containment atmosphere will be provided: the air recirculation units and the containment spray system. Each of these systems will provide sufficient steam-condensing capacity to avoid containment vessel overstress and to remove that portion of the residual heat and chemical reaction heat released to the containment.

2.4.17 CHEMICAL REACTIONS (2)

Provisions shall be included to limit the extent and the credible consequences of chemical reactions that could cause or materially augment the release of hazardous amounts of fission products from the facility.

The most severe chemical reaction is the zirconium-water reaction. Such a reaction could result if core cooling were lost for a significant time because of an uncontrolled loss of coolant from the reactor coolant system, as analyzed in Section 12.

The consequences of the zirconium-water reaction are two additional energy sources:

a) The exothermic heat of reaction stored in the core which would be released when the core is quenched, and

b) The H_2-O_2 reaction energy released to the containment. The zirconium-water reaction is limited by the same safeguard that cools the core following a loss-of-coolant accident; i.e., the safety injection system.

The safety injection system will consist of four accumulator tanks, three high head, low flow safety injection pumps, and two high flow, low head pumps of the residual heat removal system. The safety injection pumps discharge into two hot and two cold legs of the reactor coolant loops. The accumulator tanks and the residual heat removal pumps flood the core from the bottom through connections to the loop cold legs. The flow and head of the accumulator tanks and the pumps and the supply of water will be sized to give adequate core protection for the full range of break sizes. In addition, the charging pumps of the Chemical and Volume Control System will be available to augment the flow of the safety injection systems, should the reactor coolant pressure remain high.

In the case of a hypothetical accident (a double-ended break of a reactor coolant pipe) with limited safeguards operations from the emergency power supply, the core would be quickly reflooded to prevent fuel clad melting and limit metal water reaction to an insignificant amount. This ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. Limited safeguards operation here includes the accumulator tanks, two high head safety injection pumps, one residual heat removal pump, two service water pumps, three air recirculation cooling units and one containment spray pump. This complement of safeguards components is capable of operation on power supplied by two of the three emergency generators.

2.4.18 CONTAINMENT INTEGRITY (17)

The containment structure, including access openings and penetrations, shall be designed and fabricated to accommodate without failure credible transients of pressure and temperature. These transients shall be analyzed with allowance for appropriate operating and failure modes of engineered safeguards. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards shall be provided for that part as necessary to protect the health and safety of the public, in case of an accidental rupture in that part of the system.

The containment structure and any components or systems exterior to the containment structure which are in effect part of the containment system will be designed to withstand pressure loads and temperature gradients resulting from the most severe loss of coolant accident, acting in combination with dead loads and loads resulting from the maximum seismic or wind forces characteristic of the site.

The methods used in deriving the component loads are given in Chapter 5.0. The following general criteria are followed to assure conservatism in computing the required structural load capacity:

a) In calculating the containment pressure, rupture sizes up to and including a double-ended severance of a reactor coolant pipe are considered.

b) In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated. Contingent mechanical or electrical failures are assumed to disable one of the emergency generators, one of the five fan-cooler units and one of the two containment spray units. Equipment which can be run from emergency power is described in Chapter 8.

c) The pressure and temperature loadings obtained by analyzing these cases, when combined with operating loads and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening or penetrations.

2.4.19 CONTAINMENT COOLING (18)

Provision shall be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under accident conditions. If active heat dissipation systems are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems shall be provided, preferably of different principles.

To assure integrity of the containment following the hypothetical loss-of-coolant accident with no active quenching systems (safety injection into the reactor cavity), any three of the five installed fan cooler units and one spray pump, or five fan coolers units, or two spray pumps must be placed in operation for removal of residual heat.

The containment spray system will be an independent backup to the fan cooler units with the two spray pumps having the equivalent heat removal capacity of the five fan cooler units. The containment spray pumps will take suction from the refueling water storage tank. When the refueling water storage tank is exhausted, the spray water will be provided by recirculating pumps from the containment sump through the residual heat removal heat exchangers.

Service water will be the coolant for the fan coolers. Operation of two of the six installed service water pumps will provide sufficient cooling water for the five fan cooler units. One of the six installed service water pumps will also provide water for the component cooling heat exchanger which will be the heat sink for cooling the residual heat exchangers during the long term cooling period following injection. These pumps will be located at the intake structure and take suction directly from the intake canal and pump water to the component cooling heat exchangers and fan cooler units.

Electrical power for the fan motors, the service water pumps, the component cooling water pumps and spray pumps or recirculating pumps will be provided from the normal station auxiliary supply. If auxiliary power is not available, the on-site emergency power generator units will supply power as described in Chapter 8.

2.4.20 CONTAINMENT ISOLATION (22)

A reliable containment isolation system shall be provided where necessary to assure containment integrity.

Piping penetrating the containment will be designed for pressures at least equal to the containment design pressure. Containment isolation valves will be provided as necessary in lines penetrating the containment to prevent release of radioactivity. Six classes of penetrations provide at least two barriers between the containment atmosphere and the environs outside the containment. This design is such that failure of one valve to close will not prevent isolation. No manual operation will be required for immediate isolation of the containment. More details on the containment isolation system and the isolation valve seal water system are given in Chapter 6.

All remotely operated and automatically tripped isolation valves will be provided with control switches and position indication in the control room. Automatic trip isolation valves will be designed to fail in the closed position upon loss of actuating signal. Trip valves will close automatically upon a containment isolation signal derived from redundant containment pressure monitoring channels.

2.4.21 CONTAINMENT LEAKAGE (19)

The containment shall be designed so that its maximum integrated leakage under accident conditions shall meet the site exposure criteria set forth in 10 CFR 100.

The combination of continuously pressurized double penetrations and weld channels and the isolation valve seal water system assure a virtually leak-tight containment and hence assure that off-site inhalation exposures to the public will be well below 10 CFR 100 guidelines for the hypothetical loss-of-coolant accident even assuming complete core meltdown. The spray system, utilizing borated water containing sodium thiosulphate, will provide assurance that adequate time is available for operation of the isolation valve seal water system. With a containment leak-rate of 0.1 per cent per day the two-hour exposure at the minimum exclusion distance will be less than 10 CFR 100 guidelines.

The preoperational leak rate tests will include an integrated leak rate test of the containment and sensitive leak rate test of the pressurized penetrations and weld channels. The leak rate test will be an integrated leak rate test at 47 psig as a gross check against the unlikely possibility of leak through unclosed openings remaining from construction. During this test, the weld channels and penetrations will be open to the atmosphere. A sensitive leak rate test will be performed by pressurizing the weld channels and double penetrations at 54 psig. The test will be conducted with the containment at atmospheric pressure. These tests will demonstrate

the integrity of each of the double leakage barriers provided by the weld channels and penetrations and the overall integrity of the containment. The expected leakage rate to be demonstrated for the containment integrated leak test is less than 0.1 per cent of the containment free volume per day.

During operation, the double penetrations and weld channels will be continuously pressurized to provide a continuous and very sensitive and accurate means of monitoring leakage status. With these provisions, there should be no need to perform integrated leak rate tests of the containment building unless major maintenance or modification of the containment are made.

2.4.22 ACCESS PROVISIONS (13)

The facility shall be provided with adequate radiation protection to permit access, even under accident conditions, to equipment as necessary to maintain the facility in a safe condition.

During normal conditions, radiation levels do not restrict access to the operating areas of the plant outside the containment. Access to the containment during normal operations is discussed in Chapter 5. Security controls are provided to preclude unauthorized persons from gaining access to any part of the plant.

Sufficient shielding, distance, and containment integrity will be provided so that under accident conditions, radiation levels will allow access to and egress from the site and permit operators to leave the control room in order to attend necessary equipment. The permissible time out of the control room will depend on the time elapsed following the accident and the distance from the containment and will be limited so that the maximum potential whole body dose for the course of an accident will not exceed 25 rem.

2.4.23 EFFLUENT RELEASE (26)

Where environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate holdup capacity shall be provided for retention of gaseous, liquid or solid effluents.

Gaseous, liquid and solid waste disposal facilities will be designed so that discharge of effluents and off-site shipments shall be in accordance with applicable governmental regulations.

Unfavorable environmental conditions are not expected to place any restrictions on the normal release of operational radioactive effluents to the environment. Radioactive fluids entering the Waste Disposal System will be collected in sumps and tanks until the course of subsequent treatment is determined by analysis, as detailed in Section 11.

Radioactive gases will be pumped by compressors through a manifold to one of the waste gas storage tanks where they will be held for a suitable period of time for decay. Tanks will be provided for normal operation - filling, isolation for decay and discharge. Additional tanks will be provided to accommodate unexpected plant operations such as cold or hot shutdown which might produce additional waste gases. During normal operation, gases will be discharged at a controlled rate from these tanks through the monitored plant vent. Automatic features are incorporated to preclude releases in excess of 10 CFR 20 limits. If unfavorable weather conditions persist for long periods during the venting, discharge of radioactive gases may be reduced if necessary.

All solid wastes will be placed in suitable containers and stored on-site until shipment off-site for ultimate disposal.

Liquid wastes will be processed through an evaporator to remove most of the radioactive material. The spent resins from the demineralizers and the concentrates from the evaporators will be drummed and stored on-site until shipment off-site for permanent disposal. The processed water, from which most of the radioactive material has been removed, can be reused or discharged. Discharge will be via a monitored and automatically controlled line into the condenser discharge canal. Water discharge to the river will not exceed to 10 CFR 20 limits for drinking water.

2.4.24 FUEL AND WASTE FACILITIES (24 & 25)

Fuel and waste storage and handling systems shall be designed and operated in such a manner that credible accidental release of radioactivity will not exceed the limits set forth in 10 CFR 100. The fuel handling and storage facilities shall be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions and credible accident conditions.

All fuel storage and waste handling facilities will be contained and equipment design will be such that accidental releases of radioactivity directly to the environment are not possible. Refer to Sections 9 and 11, respectively.

During the refueling of the reactor, all operations will be conducted with the spent fuel under water (see Section 9). This provides visual control of the operation at all times and also maintains low radiation levels (less than 50 mr/hr) throughout the operation. The borated refueling water assures subcriticality at all times and also provides adequate cooling for the spent fuel during transfer. Spent fuel will be taken from the reactor and transferred to the refueling canal and placed in the fuel transfer system. Rod cluster control assembly transfer from a spent fuel assembly to a new fuel assembly will be accomplished prior to transferring the spent fuel to the spent fuel storage pit.

A spent fuel storage pit will be provided for decay of spent fuel prior to shipment from the site. It will be designed to accommodate the storage of a total of one and one-third core plus the space required for shipping spent fuel. The storage pit will be filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The spent fuel will be stored in a vertical array with sufficient center-to-center distance between assemblies to ensure subcriticality ($k \leq 0.9$) even if unborated water were introduced into the pit. The storage racks will be designed so that it will be impossible to insert assemblies in other than the prescribed locations. The water level maintained in the pit will provide sufficient shielding to permit normal occupancy of the area by operating personnel. The spent fuel pit will be provided with systems to maintain water cleanliness and to indicate pit water level and to provide for heat removal from the pit water. Gamma radiation is continuously monitored and a high levels will be annunciated in the control room.

Water removed from the spent fuel storage pit must be pumped out as there are no gravity drains. Spillage or leakage of any liquids from waste handling facilities will be retained within the building by accumulation in floor drains which flow into a drain tank.

A controlled ventilation system will exhaust the atmosphere in the fuel handling building and the auxiliary building and will discharge it to the atmosphere via the plant vent. Radiation monitors will be in continuous service in these areas to actuate high-activity alarms on the control board annunciator. A radiation monitor will also take continuous air samples of the plant vent gas effluent stream to actuate alarms on the control board annunciator and to terminate discharge of ventilating air or waste gases on a high activity signal.

Postulated accidents involving the release of radioactivity in the fuel and waste storage and handling facilities are shown in Section 12 to result in exposures well within the limits of 10 CFR 100.

The reactor cavity, refueling canal and the spent fuel storage pit will be reinforced concrete structures lined with seam welded stainless steel plate. These structures will be designed to withstand the anticipated earthquake loadings as Class I structures so that the stainless steel liner should prevent leakage even in the event the reinforced concrete develops cracks. The transfer tube which will connect the refueling canal to the spent fuel pit and which will form part of the reactor containment will be provided with a valve and a blind flange which will effectively close off the transfer tube when it is not in use.

New fuel will be brought into the containment through the equipment hatch or through the spent fuel pit and fuel transfer tube. The fuel will be stored vertically in racks designed to prevent criticality even in the event the storage area should become flooded. This area will be provided with drainage to prevent water accumulation.

2.5 PLANT OPERATION

Final operating procedures for the Indian Point Unit #3 Nuclear Generating Station will be established following the completion of the detail design of the facility and prior to initial operation. The administration of operation and technical operating limits for the interests of the health and safety of the public will be defined in "Technical Specifications." Included in the technical specifications will be:

a) Safety limits and maximum safety system settings
b) Minimum conditions for operation
c) Surveillance requirements
d) Design Features
e) Administrative controls

This document will be part of the submission of the final Safety Analysis Report in support of the application for an operating license.

2.6 RESEARCH AND DEVELOPMENT REQUIREMENTS

The design of Indian Point Unit No. 3 Nuclear Station will be based upon proven concepts which have been developed and successfully applied to the design of pressurized water reactor systems. Design and development work on the Indian Point Unit No. 2 Station has been underway for over a year. Design and development to support Unit No. 2 is directly applicable to Unit No. 3. Technical information is being generated in the following areas:

- Fuel management and control management studies
- Methods to control xenon instabilities
- Critical experiments at Westinghouse Reactor Evaluation Center to substantiate design details
- Development of thiosulfate spray system
- Positive moderator reactivity coefficient
- Development of emergency core cooling system with accumulators
- Effect of blowdown forces on reactor internals
- Rod ejection analysis

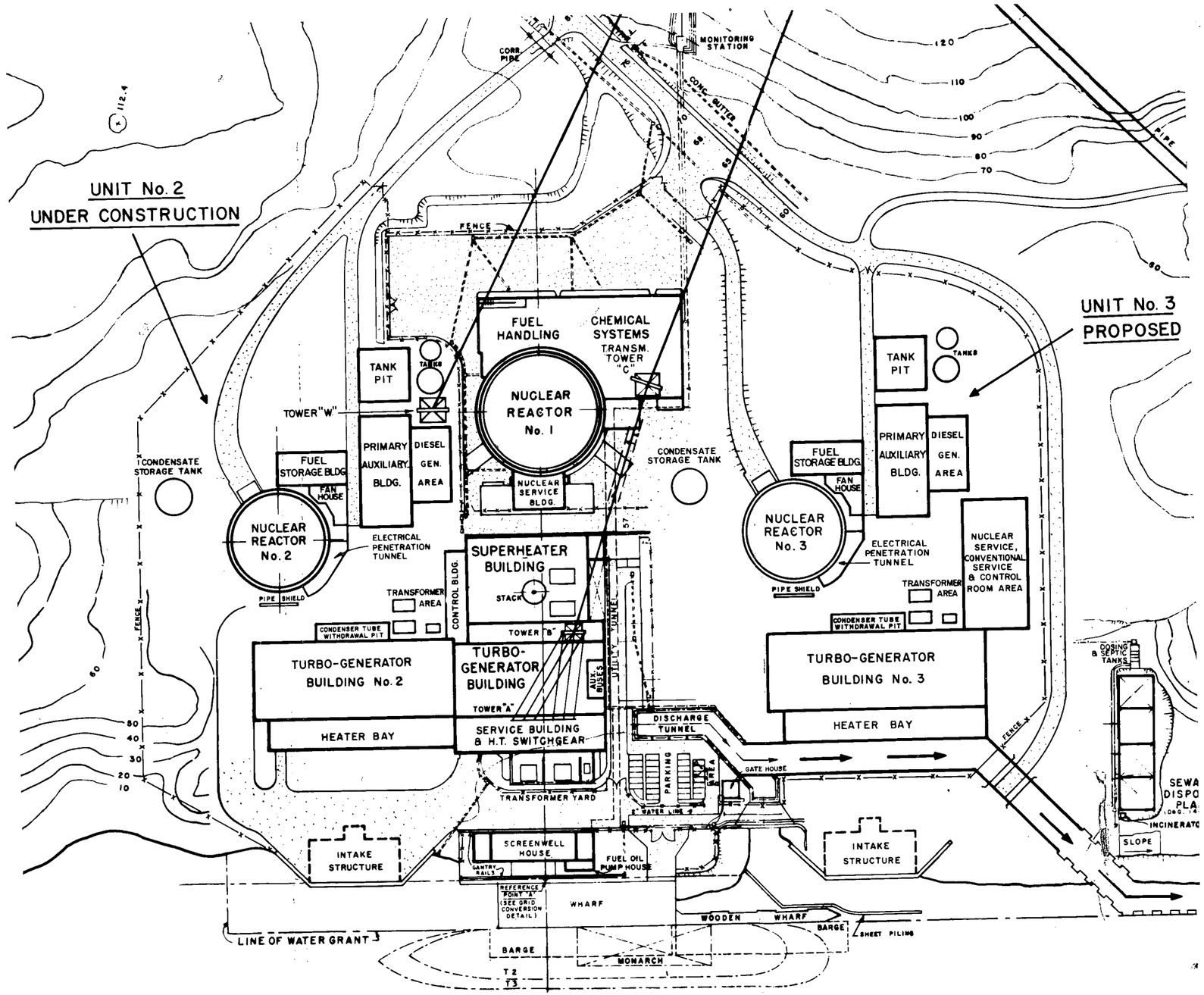
Results of work already completed under the Nuclear Safety Research and Development Program being conducted by the Atomic Energy Commission will be incorporated in the design and evaluation of applicable portions of the engineered safeguards systems. No additional research and development programs affecting plant safety are needed with respect to features of Indian Point Unit No. 3.

2.7 IDENTIFICATION OF CONTRACTORS

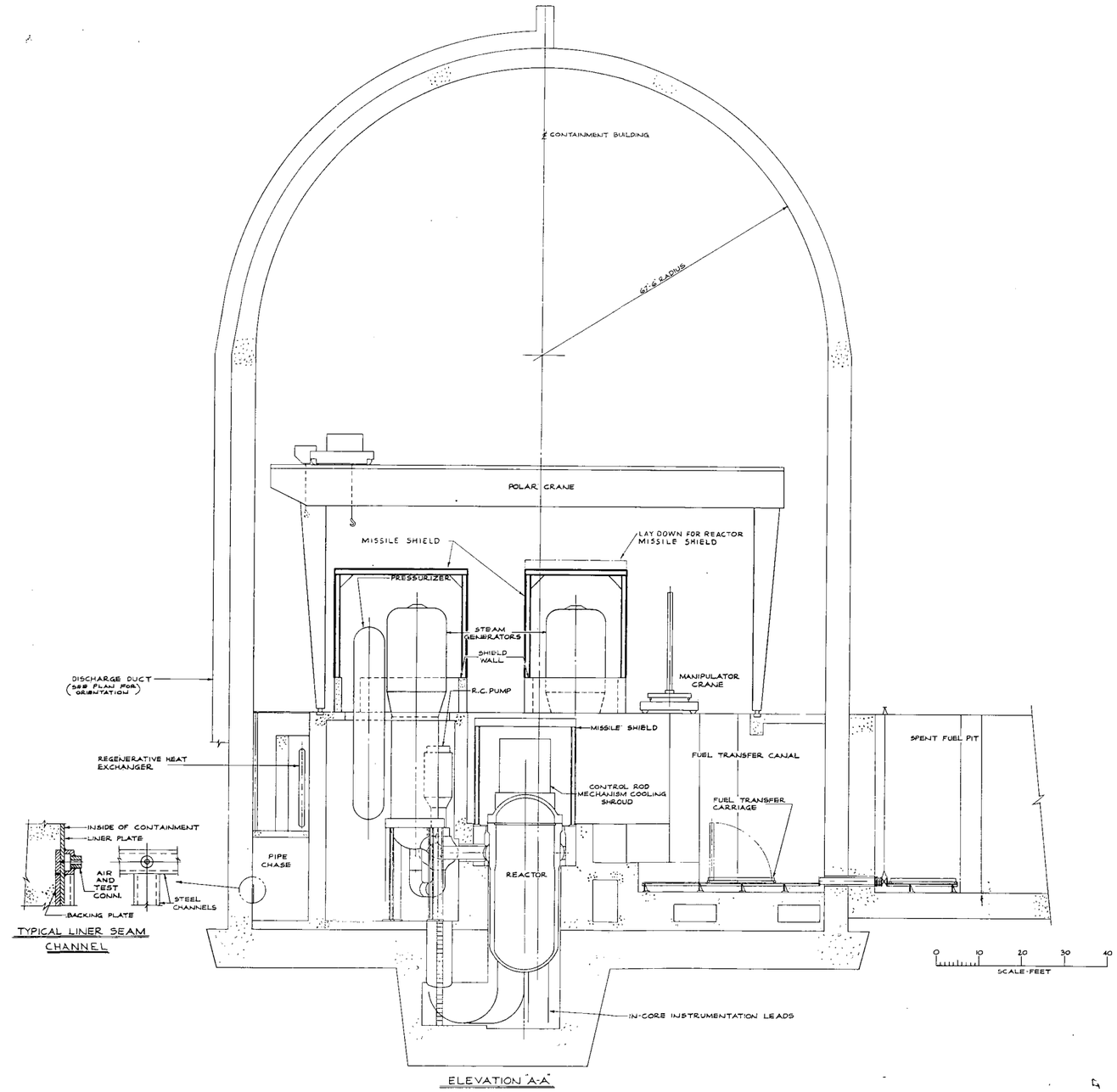
Consolidated Edison, as owners, has engaged Westinghouse Electric Corporation to design and construct the Indian Point Unit No. 3. However, irrespective of the explanation of contractual arrangements offered below, Consolidated Edison is the sole applicant for licenses as as owner and applicant is responsible for the design, construction, and operation of the Indian Point Unit No. 3.

The Indian Point Unit No. 3 will be designed and built by the Westinghouse Electric Corporation as prime contractor for Consolidated Edison. Westinghouse has undertaken to provide a complete, safe, and operable nuclear power plant ready for commercial service by June 1971. The project will be directed by Westinghouse from the offices of its Atomic Power Division in Pittsburgh, Pennsylvania and by Westinghouse representatives at the plant site during construction and plant startup. Westinghouse has engaged United Engineers and Constructors of Philadelphia, Pennsylvania to provide the design of certain portions of the plant.

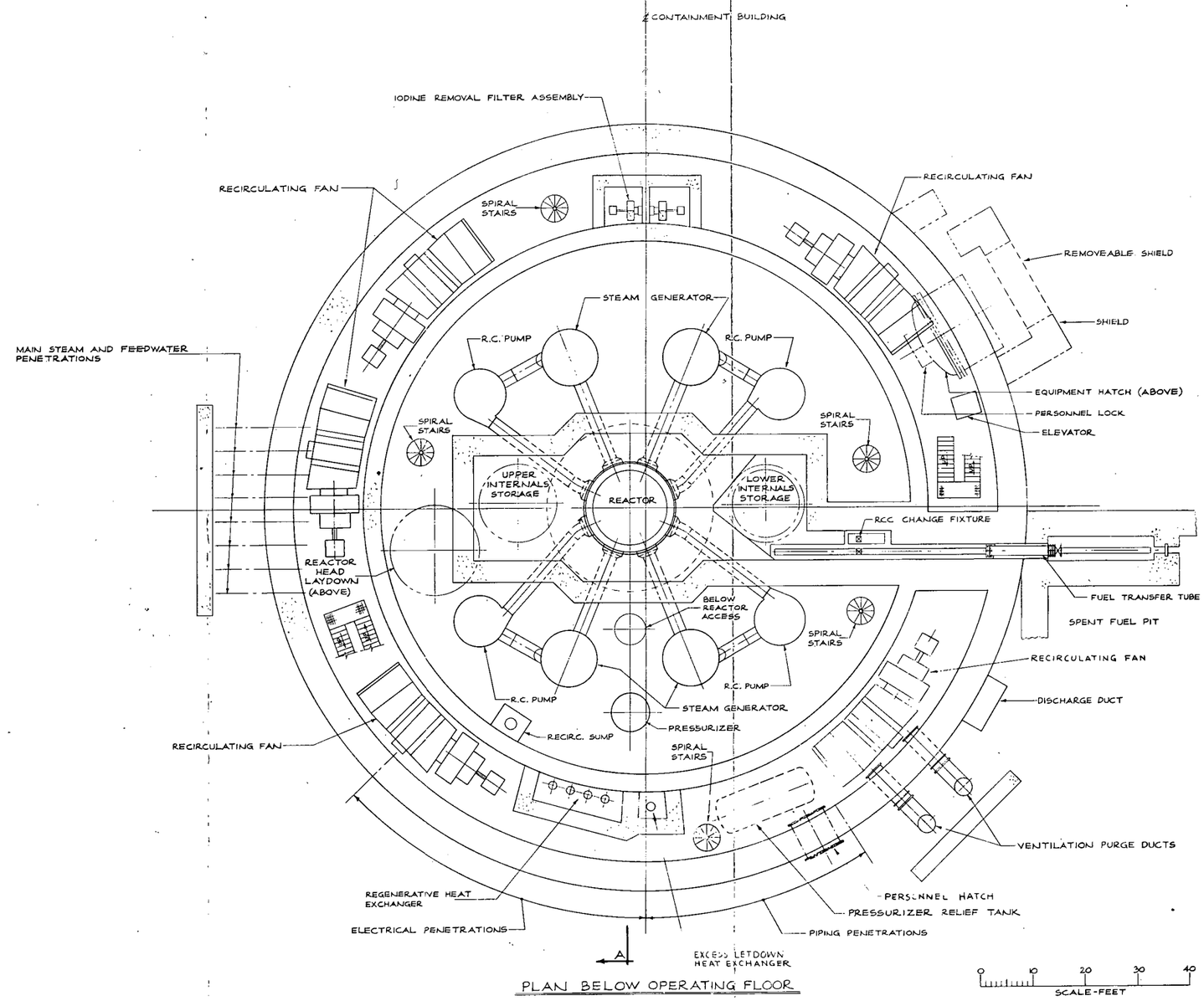
The plant will be constructed under the general direction of Westinghouse through United Engineers and Constructors which will be responsible for the management of all site construction activities and which will either perform itself or subcontract the work of construction and equipment erection. Preoperational testing of equipment and systems at the site and initial plant operation will be performed by Consolidated Edison personnel under the technical direction of Westinghouse.



SITE PLOT PLAN
FIG. 2-1



CONTAINMENT ELEVATION SECTION
FIG. 2-2



PLAN BELOW OPERATING FLOOR

CONTAINMENT PLAN SECTION
FIG. 2-3

SECTION 3

PSAR Section	PSAR Page	Remarks
3.1	3-1	Item 1 of Supplement 1 to the PSAR gives the general design criteria for the design bases described in Section 3.1. The following GDC are specifically referenced: GDC 6 page 10 GDC 7 page 11 GDC 27 page 34 GDC 28 page 35 GDC 29 page 36 GDC 30 page 37 GDC 31 page 38 GDC 32 page 39
3.1	3-1	The description of the reactor control assemblies should include part length rods which is referenced in Supplement 1 to the PSAR, Item 9. The report, WCAP 7072, gives a mechanical description and analysis of effects on the reactor core and reactor control.
3.2.1.1	3-4	Burnable poison rods have been incorporated into the design and the description is found in Supplement 1, Item 9 which references WCAP 7113. The reference report describes the reactivity changes due to burnable poison rods and the use of burnable poison rods in the fuel cycle.
3.2.1.1	3-12	The material under the heading "Xenon Induced Spatial Instability" has been superseded with the use of part length rods to control axial oscillations. The use of part length rods is described in WCAP 7072 as referenced in Supplement 1, Item 9.
3.2.2	3-41	Additional information on the thermal and hydraulic characteristic design factors is given in Supplement 1 to the PSAR, Item 2 (1 - 5), Item 10 (A - 3.0), and Item 10 (A - 4.0).
3.2.3.2	3-74	Additional information on fuel clad stress and strain is given in Supplement 1, Item 10 (A - 5.0), Item 10 (A - 6.0) and Item 10 (A - 7.0).