1.3 COMPARISON TABLES

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

This subsection highlights the historical principal design features of the plant and compares its major features with other boiling water reactor facilities. The design of this facility was based on proven technology obtained during the development, design, construction, and operation of boiling water reactors of similar types.

The data, performance, characteristics, and other information presented here represent the then current Susquehanna Steam Electric Station design as it compared to similar designs available at that time. To maintain this original design comparison, these tables will not be revised to reflect current plant design, other than the previous addition of the fifth ("E") emergency diesel generator to Tables 1.3-6 and 1.3-7.

1.3.1.1 Nuclear Steam Supply System Design Characteristics

Table 1.3-1 summarizes the design and operating characteristics for the nuclear steam supply systems. Parameters are related to rated power output for a single plant unless otherwise noted.

1.3.1.2 Power Conversion System Design Characteristics

Table 1.3-2 compares the power conversion system design characteristics.

1.3.1.3 Engineered Safety Features Design Characteristics

Table 1.3-3 compares the engineered safety features design characteristics.

1.3.1.4 Containment Design Characteristics

Table 1.3-4 compares the containment design characteristics.

1.3.1.5 Radioactive Waste Management Systems Design Characteristics

Table 1.3-5 compares the radioactive waste management design characteristics.

1.3.1.6 Structural Design Characteristics

Table 1.3-6 compares the structural design characteristics.

1.3.1.7 Instrumentation and Electrical Systems Design Characteristics

Table 1.3-7 compares the instrumentation and electrical systems design characteristics.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

All of the significant changes that have been made in the facility design between submission of the last PSAR revision and Revision 0 of the FSAR are listed in Table 1.3-8. Each item in Table 1.3-8 is cross-referenced to the appropriate portion of the FSAR which describes the changes and the bases for them.

Table 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

(Parameters are for rated power output for a single plant unless otherwise noted. These parameters were current at issuance of the SSES Operating License and are being maintained for historical reference.)

	SSES BWR 4 251-764	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 28-560**	**GESSAR BWR 6 238-732**
THERMAL AND HYDRAULIC DESIGN				
Rated power, MWt	3293	2436	2436	3579
Design power, MWt (ECCS design basis)	3439	2550	2550	3758
Steam flow rate, lb. hr.	13.48×10^6	10.03×10^6	10.477×10^6	15.396×10^6
Core coolant flow rate, lb/hr.	100.0×10^6	78.5×10^6	78.5×10^6	105.0×10^6
Feedwater flow rate, lb/hr.	13.574×10^6	10.445×10^6	10.477×10^6	15.358
System pressure, nominal in steam dome, psia	1020	1020	1020	1040
Average power density, KW/liter	48.7	51.2	50.51	56.0
Maximum thermal output, KW/ft.	13.4	13.4	13.4	13.4
Average thermal output, KW/ft.	5.34	7.11	5.45	6.04
Maximum heat flux, Btu/hr-ft^2	361,000	428,300	354,000	354,300
Average heat flux, Btu/hr-ft^2	144,100	164,700	143,900	159,600
Maximum UO_2 temperature, °F	3330	4380	3325	3337
Average volumetric fuel temperature, °F	1100	1100	1100	1100
Average cladding surface temperature, °F	558	558	558	558
Minimum critical power ratio (MCPR) * For Hatch minimum critical heat flux (MCHFR) ws used.	1.23	1.9*	1.21	1.24
Coolant enthalpy at core inlet, Btu/lb	521.8	526.2	527.4	527.9
Core maximum exit voids within assemblies	76	79	75	76
Core average exit quality, % steam	13.2	12.9	13.6	14.9
Feedwater temperature, °F	383	387.4	420	420
THERMAL AND HYDRAULIC DESIGN				
Design Power Peaking Factor:				
Maximum relative assembly power	1.40	1.40	1.40	1.40
Local peaking factor	1.15	1.24	1.24	1.13
Axial peaking factor	1.40	1.50	1.40	1.40
Total peaking factor	2.51	2.60	2.43	2.22
NUCLEAR DESIGN (First Core)				
Water/$U0_2$ volume ratio (cold)	2.80	2.53	2.41	2.70
Reactivity with strongest control rod K_{eff}	F0.99	F0.99	F0.99	F0.99
Moderate void coefficient:				
Hot, no voids $\Delta k/k$ - % void	-1.0×10^{-3}	-1.0×10^{-3}	-1.0×10^{-3}	-0.3×10^{-3}
At rated output, $\Delta k/k$ - % void	-1.7×10^{-3}	-1.6×10^{-3}	1.6×10^{-3}	-1.0×10^{-5}
Fuel temperature doppler coefficient:				
At 68 °F, $\Delta k/k$ – °F fuel	-1.2×10^{-5}	1.3×10^{-5}	-1.3×10^{-5}	-1.6×10^{-5}

Table 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

(Parameters are for rated power output for a single plant unless otherwise noted. These parameters were current at issuance of the SSES Operating License and are being maintained for historical reference.)

	SSES BWR 4 251-764	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 28-560**	**GESSAR BWR 6 238-732**
Hot, no voids, Δk/k °F fuel	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	-1.3×10^{-5}
At rated output, Δk/k,°F fuel	-1.2×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}	-1.2×10^{-5}
Initial average U-235 enrichment wt. %	1.88	2.23	1.90	1.90
Fuel average discharge exposure, MWd/short ton	16,200	19,000	15,053	13,000
CORE MECHANICAL DESIGN (First Core)				
Fuel Assembly:				
Number of fuel assemblies	764	560	560	732
Fuel rod array	8 x 8	7 x 7	8 x 8	8 x 8
Overall dimensions, in.	176	176	176	176
Weight of $U0_2$ per assembly lb. (pellet type)	458 (chamfered)	490.4 (undished) 483.4 (dished)	465.15	472 (chamfered)
Weight of fuel assembly, lb.	600	681 (undished) 675 (dished)	698	
Fuel Rods:				
Number per fuel assembly	62	49	63	63
Outside diameter, in	0.483	0.563	0.493	0.493
Cladding thickness, in	0.032	0.032	0.034	0.034
Gap, pellet to cladding, in	0.0045	0.006	0.0045	0.009
Length of gas plenum, in	10	16	14	12
Cladding material* *Free-standing loaded tubes	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel Pellets:				
Material	$U0_2$	$U0_2$	$U0_2$	$U0_2$
Density, % of theoretical	95	95	95	94
Diameter, in	0.410	0.487	0.416	0.416
Length, in	0.410	0.5	0.420	0.420
Fuel Channel:				
Overall dimension, length, in	166.9	166.9	166.9	
Thickness, in	0.080	0.080	0.100	0.120
Cross section dimensions, in	5.48 x 5.48	5.44 x 5.44	5.48 x 5.48	5.52 x 5.52
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core Assembly:				
Fuel Weight as UO_2, lb	349,000	272,850	250,538	345,500
Core diameter, (equivalent), in	187.1	160.2	160.2	
Core height (active fuel) in	150	144	146	148

Table 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

(Parameters are for rated power output for a single plant unless otherwise noted. These parameters were current at issuance of the SSES Operating License and are being maintained for historical reference.)

	SSES BWR 4 251-764	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 28-560**	**GESSAR BWR 6 238-732**
Reactor Control System:				
Method of variation of reactor power	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow
Number of movable control rods	185	137	137	177
Shape of movable control rods	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of movable control rods	12.0	12.0	12.0	12.0
Control material in movable rods	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes
Type of control rod drives	Bottom entry locking piston	Bottom entry locking piston	Bottom entry locking piston	Bottom entry locking piston
Type of temporary reactivity control for initial core	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods
Incore Neutron Instrumentation:				
Number of incore neutron detectors (fixed)	172	124	124	164
Number of incore detector assemblies	43	31	31	41
Number of detectors per assembly	4	4	4	4
Number of flux mapping neutron detectors	5	4	4	
Range (and number) of detectors:				
Source range monitor	Source to 0.001% power (4)	Source to 0.001% power (4)	Source to 0.001% power (4)	Source to 0.001% power
Intermediate range monitor	0.001% to 10% power (8)	0.001% to 10% power (8)	0.001% to 10% power (8)	0.001% to 10% power
Local power range monitor	5% to 125% power (172)	5% to 125% power (124)	5% to 125% power (124)	5% to 125% power
Average power range monitor *Channels of monitors from LPRM detectors.	5% to 125% power (6)*	2.5% to 125% power (6)*	2.5% to 125% power (6)*	2.5% to 125% power
Number and types of incore neutron sources	7 Sb-Be	5 Sb-Be	5 Sb-Be	
REACTOR VESSEL DESIGN				
Material	Carbon steel/Stainless Clad	Carbon steel/Stainless Clad	Carbon steel/Stainless Clad	Carbon steel/Stainless Clad
Design pressure, psig	1250	1265	1250	1250
Design temperature, °F	575	575	575	575
Inside diameter, ft-in.	20-11	18-2	18-2	19-10
Inside height, ft-in.	72-11	69-4	69-4	70-10
Minimum base metal thickness(cylindrical section) in	6.19	5.53	5.375	5.70
Minimum cladding thicknesses, in	1/8	1/8	1/8	1/8
REACTOR COOLANT RECIRCULATION DESIGN				
Number of recirculation loops	2	2	2	2
Design pressure:				

Table 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

(Parameters are for rated power output for a single plant unless otherwise noted. These parameters were current at issuance of the SSES Operating License and are being maintained for historical reference.)

	SSES BWR 4 251-764	**HATCH 1 BWR 4 218-560**	**ZIMMER BWR 5 28-560**	**GESSAR BWR 6 238-732**
Inlet leg, psig	1250	1148	1250	1250
Outlet leg, psig * Pump and discharge piping to and including discharge block valves. ** Discharge piping from discharge block valve to vessel	1500	1274	1675*;1575**	1675*;1575**
Design temperature, °F	575	562	575	575
Pipe diameter, in	28	28	20	22/24
Pipe material, ANSI	304/316	304/316	304/316	304
Recirculation pump flow rate, gpm	45,200	42,200	33,880	35,400
Number of jet pumps in reactor	20	20	20	20
MAIN STEAMLINES				
Number of streamlines	4	4	4	4
Design pressure, psig	1250	1146	1250	1250
Design temperature, °F	575	563	575	575
Pipe diameter, in	26	24	24	26
Pipe material	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel

Table 1.3-2

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

(Parameters are for rated power output for a single plant unless otherwise noted. These parameters were current at issuance of the SSES Operating License and are being maintained for historical reference.)

	SSES BWR 4 251-764	HATCH BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
TURBINE GENERATOR (See Sections 10.2 and 10.4)				
Rated power, MWt	3293	2550	2550	*
Rated power, MWe (gross)	1085	813	883	*
Generator Speed, RPM	1800	1800	1800	*
Rated steam flow, lb/hr	13.4×10^6	10.48×10^6	11.0×10^6	*
Inlet pressure, psig	965	950	950	*
STEAM BYPASS SYSTEM (See Section 10.4.4)				
Capacity, % design steam flow	25	25	25	*
MAIN CONDENSER (See Section 10.4.1)				
Heat removal capacity, Btu/hr	7890×10^6	5720×10^6	7053×10^6	*
CIRCULATING WATER SYSTEM (See section 10.4.5)				
Number of pumps	4	2	3	*
Flow rate, gpm/pump	112,000	185,000	150,000	*
CONDENSATE AND FEEDWATER SYSTEM				
Design flow rate, lb/hr	13.44×10^6	10.096×10^6	10.971×10^6	*
Number of condensate pumps	4	3	3	*
Number of condensate booster pumps	None	3	3	*
Number of feedwater pumps	3	2	2	*
Number of feedwater booster pumps	None	None	None	*
Condensate pump drive	AC Power	AC Power	AC Power	*
Booster pump drive	NA	AC Power	AC Power	*
Feedwater pump drive	Turbine	Turbine	Turbine	*
Feedwater booster pump drive	NA	NA	NA	*
*See applicant's SAR				

Table 1.3-3

COMPARISON OF ENGINEERED SAFETY FEATURES DESIGN CHARACTERISTICS

Security-Related Information

Table Withheld Under 10 CFR 2.390

Table 1.3-4
COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 1.3-5

RADIOACTIVE WASTE MANAGEMENT SYSTEMS DESIGN CHARACTERISTICS

Security-Related Information

Table Withheld Under 10 CFR 2.390

Table 1.3-6

COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS

(Parameters are for rated power output for a single plant unless otherwise noted. These parameters were current at issuance of the SSES Operating License and are being maintained for historical reference.)

	SSES BWR-4 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560	GESSAR BWR 6 238-732
SEISMIC DESIGN* (See Section 3.7)				
Operating Basis Earthquake				
- horizontal g	0.05	0.08	0.10	0.15
- vertical g	0.033	0.05	0.07	----
Safe shutdown earthquake				
- horizontal g	0.10	0.15	0.20	0.30
- vertical g	0.067	0.10	0.14	----
WIND DESIGN (See Section 3.3)				
Maximum sustained – mph	80	105	90	130
TORNADOS (See Section 3.3)				
Translational – mph	60	60	60	70
Tangential – mph	300	300	300	290

*Some of the tabluated values differ for the design of the Diesel Generator ‘E” Facility.

Table 1.3-7

COMPARISON OF ELECTRICAL POWER SYSTEM DESIGN CHARACTERISTICS

Security-Related Information
Table Withheld Under 10 CFR 2.390

TABLE 1.3-8

SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR*

ITEM	CHANGE	REASON FOR CHANGE	FSAR PORTION IN WHICH CHANGE IS DISCUSSED
Recirculation flow measurement	The recirculation flow measurement design was changed from a flow element to an elbow-tap type.	To improve flow measurement accuracy.	7.3.1, 7.6.1
Recirculation system	The pressure interlock for RHR shutdown mode was changed.	NRC Requirement for diversity.	7.3.1, 7.6.1
Nuclear fuel	The number of fuel pins in each fuel bundle has been changed from 7 x 7 to 8 x 8.	Improved fuel performance by increasing safety margins.	4.2
Nuclear boiler	An additional test mode was added for closing MSIV's one at a time to 90% of full open in the fast mode (close in slow mode already existed).	Verifies that the spring force on the valves will cause them to close under loss-of-air conditions.	5.4
Main steam line isolation	A main condenser low vacuum initiation of the main steam line isolation was added.	NRC requirement	7.3.1
Main steam line isolation	Reactor isolation was deleted for high water level initiation actuation.	To provide improved plant availability.	5.4
Main steam line drain system	A main steam line drain system was improved.	Prevent accumulation of condensate in an idle line outboard of MSLIV.	5.4
Feedwater sparger	The thermal sleeve was changed to provide improved design of sparger to nozzle.	To eliminate vibration, failure, and leakage.	5.3
Standby liquid control (SLC) system	Interlocks on the SLC system were revised.	To prevent inadvertent boron injection during system testing.	9.3.5 and 7.4.1

TABLE 1.3-8

SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR*

ITEM	CHANGE	REASON FOR CHANGE	FSAR PORTION IN WHICH CHANGE IS DISCUSSED
RCIC & HPCI steam supply	A warmup bypass line and valve was added.	Permits pressurizing and pre-warming of the steam supply line downstream to the turbine during reactor vessel heatup.	5.4 and 6.3
RCIC & HPCI vacuum breaker system	A vacuum breaker system was added to the turbine exhaust line into the suppression pool.	To prevent backup of water in the pipe and consequential high dynamic pipe loads and reactions.	5.4 and 6.3
RCIC & HPCI system	Each component has been made capable of functional testing.	Improved testability	5.4 and 6.3
Automatic depressurization system (ADS)	The interlocks on the automatic depressurization system were revised.	To meet IEEE-279 requirements.	7.3.1
RPV code	The RPV was partially updated to ASME 1971 code and Summer 1971 addenda.	Update to applicable code as much as practical.	5.2
Level instrumentation	The RPV level instrumentation was revised to eliminate Yarway columns and replace them with a conventional condensing chamber type; also, separation and redundancy features were added.	Improve ECCS separation per IEEE 279 and improve reliability.	7.3.1
Leak detection system	The leak detection system was revised to upgrade the capability.	To meet IEEE-279 requirements.	7.6.1
Reactor vibration monitoring	A confirmatory vibration monitoring test was added.	NRC requirement	14.2

TABLE 1.3-8

SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR*

ITEM	CHANGE	REASON FOR CHANGE	FSAR PORTION IN WHICH CHANGE IS DISCUSSED
Primary Containment Concrete	Delineation of compressive strengths for pozzolan vs. non-pozzlan Type II Portland cements.	Update to reflect current engineering design requirements.	3.8B
RPV Insulation	Correct the RPV Insulation Description	Revised support beams on as-build RPV Insulation Panels	5.3.3.1.4
Safety Related Conduits & Trays	Correct separation statements for conduits and trays.	Question 7.4 of Amend. #5 of PSAR (Revised per requirement of Reg. Guide 1.75 – 1974).	3.12
Tornado Loading	Revised Tornado Loading combinations.	To reflect latest NRC recommendations in the Standard Review Plan.	3.3

*NOTE: Design changes listed are only those which have occurred between the last SSES PSAR Amendment and Revision 0 of the FSAR. The NRC has been notified of all other design changes prior to the last PSAR amendment by previous amendments to the PSAR.