1.3 COMPARISONS

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

Table 1.3.1 presents a summary of the characteristics of Waterford 3 for Cycle 1. The table presents comparative data for San Onofre Units 2 and 3; Arkansas Nuclear One, Unit 2; and St. Lucie Unit 1.

The San Onofre Units 2 and 3, and Arkansas Nuclear One, Unit 2 designs were selected for comparison because of the basic similarity of the reactor cores and the Reactor Coolant Systems. Also they are well advanced in terms of licensing relative to Waterford 3. St. Lucie Unit 1 was selected because of the basic similarity in the containment design.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

1.3.2.1 General

This section contains a discussion of all significant changes that have been made in the Waterford 3 design since submittal of the PSAR until the docketing of the FSAR. Changes considered as significant include changes in design bases or criteria for safety-related structures, systems or components, plant arrangement, mode of system operation, type of equipment, or gross changes in component or system capacity. In general changes have been made to further increase the safety margin and operating flexibility of Waterford 3.

1.3.2.2 Site Characteristics

Additional site studies and field tests have been made since the submittal of the PSAR. No significant site characteristic changes have been brought to light that would require a design change.

1.3.2.3 Design Criteria

1.3.2.3.1 NRC Regulatory Guides

Many design changes are the result of the evolution of NRC's interpretation of the safety requirements to comply with the General Design Criteria of 10CFR50, Appendix A. This evolution has resulted in the promulgation of many Regulatory Guides which were not addressed in the PSAR. The intent of these new guides have been evaluated against the plant design and changes have been initiated where practical to bring Waterford 3 into general compliance.

1.3.2.3.2 Design Codes

a) Containment Vessel Code Case

→(DRN 01-758)

The steel containment vessel has been designed to ASME Section III, Class "MC" rather than Class "B" requirements as stated in the PSAR. Class "B" was an earlier version of the ASME Code for containment vessels which was superseded by Class "MC" in 1971. The containment vessel design is described in Section 3.8.

←(DRN 01-758)

b) Changes in Concrete Quality Control Program

Concrete slump test frequency and maximum temperature have been changed in accordance with ASTM C-143 and ACI-305-72, respectively. The code requirements for concrete are discussed in Section 3.8

1.3.2.3.3 Pipe Break Criteria

High and moderate energy piping systems have been analyzed in accordance with NRC Branch Technical Positions APCSB 3-1 and MEB 3-1 as discussed in LP&L letter LPL3690 dated July 11, 1975. (See Section 3.6.)

1.3.2.3.4 Tornado Design Criteria

As discussed in Section 3.3, protection against multiple tornado generated missiles has been extended to systems and components required for safe shutdown.

1.3.2.4 Reactor

Changes to fuel rod design have been made in order to reduce fuel densification problems. The number and design of fuel spacer grids has been changed to enhance fuel performances under seismic loading. Fuel design is described in Section 4.2.

1.3.2.5 Reactor Structures

1.3.2.5.1 Reactor Vessel Grillage Foundation

The reactor vessel grillage has been modified to incorporate a positive system of restraints to prevent upward motion of the reactor vessel following a LOCA. This Support System is discussed in Section 5.4.

1.3.2.5.2 Reactor Vessel Cavity

The design of the reactor vessel cavity has been modified to provide a reduction in neutron streaming from the cavity, and activation of Ar-40 in the containment atmosphere. The reactor vessel cavity is discussed in Sections 3.8 and 6.2.

1.3.2.6 Engineered Safety Features

1.3.2.6.1 Hot Leg Injection Capability

Provision has been made in the Safety Injection System to permit introduction of safety injection fluid during recirculation through both the hot and cold legs, thereby ensuring adequate long term post-LOCA cooling. The Safety Injection System is described in Section 6.3.

1.3.2.6.2 Containment Ductwork Declassification

The containment ductwork ring header has been changed from Safety Class 2, seismic Category I to NNS, non-seismic Category I. The ring header is not required for the Containment Cooling System to perform its design function.

This duct system is discussed in Section 6.2.

1.3.2.6.3 Containment Vessel Installation, Testing and Inspection

Several changes have been made to the installation and testing procedures for the containment vessel. These changes include upgrading of containment vessel inner surface coating and cleaning requirements, deletion of radiography of randomly selected welds (15 percent) following postweld heat treatment, revision to welder qualification test record retention procedures, and acceptance of gas metal arc welding processes. The containment vessel is discussed in Section 6.2 and 3.8.

1.3.2.7 Instrumentation and Control

1.3.2.7.1 Reactor Protection System (RPS)

The Reactor Protection System (RPS) described in the PSAR has been expanded and some portions modified in order to provide automatic protection against axial xenon oscillations and to implement design improvements.

a) The following changes were made to meet the requirement for automatic protection against axial xenon oscillations:

   1) The high local power density trip is added;

   2) The thermal margin/low pressure trip is replaced by the low DNBR trip;

   3) The core protection calculators (CPCS) are added to provide the high local power density and low DNBR trips and the thermal margin/low pressure calculator is eliminated.

b) As a consequence of the above addition of the CPCS, the following design changes are implemented:

   1) The low reactor coolant flow trip function is incorporated into the low DNBR trip;

   2) Reactor coolant flowrate is calculated by use of reactor coolant pump speed instead of being inferred by differential pressure measurements;

   3) CEA position signals are incorporated into the RPS.

c) A high logarithimic power level trip has replaced the high rate of change of power trip in order to provide improved protection against inadvertent boron dilution. The RPS also addresses the unplanned withdrawal of CEAs as the previous trip did.

The Engineered Safety Feature Actuation System (ESFAS) has been changed in the following areas:

a) The Emergency Feedwater Actuation Signal (EFAS) is added to the ESFAS.

b) Diverse signals for Containment Isolation have been added.

c) Variable setpoints for SIAS on low pressurizer pressure and Main Steam Isolation Signal on low steam generator pressure are added.

d) The group testing capability is added.

Further discussion of the RPS is found in Section 7.2.

1.3.2.7.2 Core Operation Limit Supervisory System

→

A non-safety-related Core Operating Limit Supervisory System (COLSS) has been added. The COLSS consists of sensors, algorithms implemented in the plant monitoring computer, and other equipment to monitor selected Nuclear Steam Supply System parameters and process the parameter information so that a comprehensive, on-line calculation of the margin to specified limiting conditions of operation is available at all times. The COLSS also provides the operator with an alarm so that he can maintain the reactor core within the limiting conditions of operation during steady-state operation by initiating a power reduction whenever any one of the monitored core conditions reaches its specified limiting condition of operation. This system is described further in Section 7.7.

←

1.3.2.7.3 Movable In-Core Instrument System

This system has been deleted.

1.3.2.7.4 Deletion of Containment Purge Isolation Signal (CPIS) and High Containments Radiation Input to the Plant Protection System

Since a fuel handling accident can occur only during shutdown, CPIS on high radiation should be independent of the Plant Protection System (PPS). Therefore, the PPS has been changed accordingly. This change also allows the deenergizing of the PPS for maintenance and inspection during shutdown. The PPS is discussed in Section 7.3.

1.3.2.8 Electric Power

Extensive redesign of all electrical and I&C cable trays, control boards and cabinets to meet the separation criteria of Regulatory Guide 1.75 (1/75) has been made. Compliance with RG 1.75 is discussed in Section 8.3.

1.3.2.9 Auxiliary Systems

1.3.2.9.1 High Density Spent Fuel Storage

Additional storage capacity of spent fuel has been provided for by means of high density poison racks. The design of these racks is described in Section 9.1.

1.3.2.9.2 Fire Protection System

The Fire Protection System has been modified as a result of the new NRC requirements (10CFR50, Appendix R). The Fire Protection System is discussed in Section 9.5.

1.3.2.10 Steam and Power Conversion System

There are no significant changes in the final design of the Steam and Power Conversion System from that described in the PSAR other than all volatile water treatment has been provided.

1.3.2.11 Radioactive Waste Management

Additional radwaste handling capability has been provided for by installing means to allow the use of portable solidification and demineralization systems.

1.3.2.12 Radiation Protection

Since the PSAR, the normal sampling system has been extensively rerouted and a new sampling panel with a modified equipment configuration has been introduced in the RAB design. A new sampling system has been installed for Post-Accident Sampling (P.A.S.S.) and additional shielding and sample tubing have been installed for this purpose. These systems are discussed in Subsection 12.5.3.

1.3.2.13 Conduct of Operations

Since the PSAR, extensive modifications to the Plant Security System have been made including the addition of much more sophisticated equipment, such as TV monitors and electronic card readers. The Plant Security System is summarized in Section 13.6.

A separate Administration Building has been added due to increases in the operating staff.

1.3.2.14 Initial Test and Operations

There are no significant changes in initial tests and operations affecting plant design from that described in the PSAR.

1.3.2.15 Accident Analyses

The methods used to analyze some of the accidents have been revised to take into account expansion of the RPS (see Subsection 1.3.2.7.1). In addition, there have been extensive refinements to computer codes, analytical investigations and tests to demonstrate compliance with 1OCFR50, Appendix K,. Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors.

The offsite accident doses presented in the PSAR were calculated using the atmospheric dispersion factors (X/Q) based on preliminary meteorological data. The dose analyses presented in Chapter 15 are based on X/Q values obtained from the onsite meteorological monitoring program described in Section 2.3. The onsite data demonstrates the conservatism of the X/Q values used in the PSAR accident analyses.

1.3.2.16 Technical Specifications

Details of safety limiting settings and limiting conditions of operation have changed as a result of changes enumerated in other parts of this subsection.

Subject coverage is in accordance with NRC Standard Technical Specifications as revised for Waterford 3.

1.3.2.17 Quality Assurance

The major change in the QA program since the submittal of the PSAR has been the commitment of WASH 1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants." Quality Assurance during operation is described in detail in the QA Program Manual.

TABLE 1.3-1 (Sheet 1 of 13) Revision 11-A (02/02)

→(DRN 01-758)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

←(DRN 01-758)

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Hydraulic and Thermal Design Parameters					
Rated core heat output, Mwt	3,390	4.4	3,390	2,815	2,560
Rated core heat output, Btu/hr	11,570 x 10^6	4.4	11,570 x 10^6	9,608 X 10^6	8,737 X 10^6
Heat generated in fuel, %	97.5	4.4	97.5	96.5	97.5
System pressure, nominal, psia	2,250	4.4	2,250	2,250	2,250
System pressure, minimum steady state, psia	2,200	4.4	2,200	2,200	2,200
Hot channel factors,					
Heat flux, F_q	2.35		2.35	2.35	2.85
Enthalpy rise, F_H	1.55	4.4	1.55	1.55	2.02
DNB ratio at nominal conditions	2.07 (CE-1)	4.4	2.07 (CE-1)	2.26 (W-3)	2.30 (W-3)
Coolant flow					
Total flowrate, lb/hr	148 x 10^6	4.4	148 x 10^6	120.4 x 10^6	122 x 10^6
Effective flowrate for heat transfer, lb/hr	144.2 x 10^6	4.4	144.2 x 10^6	116.2 x 10^6	117.5 x 10^6
Effective flow area for heat transfer, ft^2	54.7	4.4	54.7	44.7	53.5
Average velocity along fuel rods, ft/sec	16.4	4.4	16.4	16.4	13.6
Average mass velocity, lb/hr-ft^2	2.64 x 10^6	4.4	2.64 x 10^6	2.60 x 10^6	2.20 x 10^6
Coolant temperatures, F					
Nominal inlet	553	4.4	553	553.5	538.9
Design inlet	553	4.4	556	556 5	544
Average rise in vessel	58	4.4	58	58.5	55
Average rise in core	60	4.4	60	60.5	56
Average in core	583	4.4	583	583.75	572

TABLE 1.3-1 (Sheet 2 of 13) Revision 9 (12/97)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Hydraulic and Thermal Design Parameters (Cont.)					
Average in vessel	582	4.4	582	582.75	571.5
Nominal outlet of hot channel	642	4.4	642	652	640
Average film coefficient, Btu/hr-ft^2-F	6,160	4.4	6,160	6,170	5,300
Average film temperature difference, F	30	4.4	30	31	35
Heat transfer at 100% power					
Active heat transfer surface area, ft^2	62,000	4.4	62,000	51,000	48,400
Average heat flux, Btu/hr-ft^2	182,400	4.4	182,400	182,200	176,000
Maximum heat flux, Btu/hr/ft^2	428,000	4.4	428,000	425,800	501,300
Average thermal output, KW/ft (Fuel Rod Only)	5.34	4.4	5.34	5.34	5.94
Maximum thermal output, KW/ft (Fuel Rod Only)	12.5	4.4	12.5	12.5	17
Maximum clad surface temperature at nominal Pressure, F	657.0	4.4	657.0	657	657
Fuel center temperature, F maximum at 100% power	3,420	4.4	3,420	3,420	3,890
Core Mechanical Design Parameters					
Fuel assemblies					
Design	CEA	4.2	CEA	CEA	CEA
Rod pitch, in.	0,506	4.2	0.5063	0.5063	0.58
Cross-section dimensions, in.	7.972 x 7.972	4.2	7.972 x 7.972	7.98 x 7.98	7.98 x 7.98
Fuel weight (as UO_2), lb_m	223.9 x 10^3	4.2	223.9 x 10^3	183,640	207,200
Total weight, lb_m	310,744	4.2	314,867	256,827	271,280
Number of grids per assembly	11	4.2	11	12	8

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Core Mechanical Desiltn Parameters (Cont.)					
Fuel rods					
Number	49,580	4.2	49,500	40,716	36,896
→(DRN 01-758)					
Outside diameter, in.	0.382	4.2	0,382	0,382	0.44
Diametral gap, in.	0.007	4.2	0,007	0,007	0.0085
Clad thickness, in.	0.025	4.2	0,025	0,025	0,026
←(DRN 01-758)					
Clad material	Zircaloy-4	4.2	Zircaloy-4	Zircaloy	Zircaloy
Fuel pellets					
Material	UO_2 sintered	4.2	UO_2 sintered	UO_2 sintered	UO_2 sintered
Diameter, in.	0.325	4.2	0.325	0.325	0.3795
Length, in.	0.390	4.2	0.390	0.390	0.650
Control assemblies					
Neutron absorber	(See Table 4.2-1)	4.2	(See Table 4.2-1)	B_4 C/Ag-In-Cd	B_4 C/SS
Cladding material	Inconel 625	4.2	Inconel 625	NiCrFe alloy	NiCrFe alloy
Clad thickness	0.035	4.2	0.035	0.035	0.040
Number of assembly, full/part-length	83/8	4.2	83/8	73/8	73/8
Number of rods per assembly	4,5/5	4.2	4,5/5	5	5
Nuclear Design Data					
Structural characteristics					
Core diameter, in. (equivalent)	136	4.2	136	123	136
Core height, in. (active fuel)	150	4.2	150	150	136.7

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Nuclear Design Data (Cont.)					
H_2 O/U, Unit cell (cold)	3.35	4.3	3.35		1.63
Number of fuel assemblies	217	4.2	217	177	217
UO_2 Rods per assembly, unshimed/shimed					
Batch A	236	4.3	236	176	176
Batch B	236/220	4.3	236/220	164	164
Batch C	236/224 or 220	4.3	236/224 or 220	176/164	176/164/164
Performance characteristics loading technique	3-batch mixed central zone	4.3	3-batch mixed central zone	3-batch mixed central zone	3-batch mixed central zone
Fuel discharge burnup, MWD/MTU					
Average first cycle	12,731	4.3	12,731	12,500	12,800
Feed enrichment, wt%					
Region 1	1.87	4.3	1.87	1.93	1.93
Region 2	2.38	4.3	2.38	2.27	2.33
Region 3	2.88	4.3	2.88	2.94	2.82
Control characteristics effective multiplication (beginning of life)					
Cold, no power, clean	1,170	4.3	1,170	1,182	1.170
Hot, no power, clean	1,125	4.3	1,125	1,136	1.134
Hot, full power, Xe equilibrium	1,067	4.3	1,067	1,075	1.078
Control assemblies					
Total rod worth (hot), %	11.35	4.3	11.35	12.3	11.0

TABLE 1.3-1 (Sheet 5 of 13) Revision 9 (12/97)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Boron concentrations for criticality:					
Zero power no rods inserted, clean, ppm	832		832		
Cold/Hot	899/832	4.3	899/832	1,004/987	945/935
At power with no rods inserted, clean/equilibrium xenon, PPM	719/452	4.3	719/452	870/612	820/590
Kinetic characteristics, range over life					
Moderator temperature coefficient, p/F	See Table 4.3-4	4.3	See Table 4.3-4	-0.5×10^{-4} to -3.1×10^{-4}	0.4×10^{4} to -2.1×10^{-4}
Moderator pressure coefficient, p/psi	$+0.7 \times 10^{-6}$	4.3	$+0.7 \times 10^{-6}$	$+0.45 \times 10^{-6}$ to $+2.97 \times 10^{-6}$	$+0.49 \times 10^{-6}$ to $+2.55 \times 10^{-6}$
Moderator void coefficient, p/% Void	-0.36×10^{-3}	4.3	-0.36×10^{-3}	-0.28×10^{-3} to -1.47×10^{-3}	-0.26×10^{-3} to -1.35×10^{-3}
Doppler coefficient, p/F	-1.13×10^{-5} to -1.67×10^{-5}	4.3	-1.13×10^{-5} to 1.67×10^{-5}	-1.18×10^{-5} to -1.78×10^{-5}	-1.45×10^{5} to -1.07×10^{-5}
Reactor Coolant System-Code Requirements					
Component					
Reactor vessel	ASME III Class 1	5.2	ASME III Class 1	ASME III Class A	ASME III Class A
Steam generator					
Tube side	ASME III Class 1	5.2	ASME III Class 1	ASME III Class A	ASME III Class A
Shell side	ASME III Class 2	5.2	ASME III Class 2	ASME III Class A	ASME III Class A

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PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Nuclear Design Data (Cont.)					
Pressurizer	ASME III Class 1	5.2	ASME III Class 1	ASME III Class 1	ASME III Class A
Pressurizer relief (or quench) tank	ASME VIII Div. 1	5.4	ASME VIII Div. 1	ASME III Class C	ASME III Class C
Pressurizer safety valves	ASME III Class 1	5.2	ASME III Class 1	ASME III Class A	ASME III Class
Reactor coolant piping	ASME III Class 1	5.2	ASME III Class 1	ASME III Class 1 (USAS B31.1)	USAS B31.7 (USAS B31.1)
Principal Design Parameters of the Reactor Coolant System					
Operating pressure, psig	2,235	5.1	2,235	2,235	2,235
Reactor inlet temperature, F	553	5.1	553	553.5	539.7
Reactor outlet temperature, F	611.2	5.1	611.2	612.5	595.1
Number of loops	2	5.1	2	2	2
Design pressure, psig	2,485	5.1	2,485	2,485	2,485
Design temperature, F	650	5.1	650	650	650
Hydrostatic test pressure (cold), psig	3,110		3,110	3,110	3,110
Total coolant volume, ft^3	10,300 (without pressurizer)		10,300 (without pressurizer)	9,376	11.101
Principal Design Parameters of the Reactor Vessel					
Material	See Table 5.2-3	5.2	See Table 5.2-2	SA-533, Grade B Class 1, low alloy steel, internally clad with Type 304 austenitic SS	SA-533, Grade B Class, 1, low alloy steel, internally clad with Type 304 austenitic SS

TABLE 1.3-1 (Sheet 7 of 13) Revision 9 (12/97)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Reactor Vessel (cont'd)					
Design pressure, psig	2,485	5.4	2,485	2,485	2,485
Design temperature, F	650	5.4	650	650	650
Operating pressure, psig	2,235	5.4	2,235	2,235	2,235
Inside diameter of shell, in.	172	5.4	172	157	172
Outside diameter across nozzles, in.	253		253	238	253
Overall height of vessel and enclosure head, ft-in. to top of CEDM nozzle	43-6-1/2	5.4	43-6-1/2	43-4-1/6	41-11-3/4
Minimum clad thickness, in.	1/8	5.4	1/8	1/8	5/16
Principal Design Parameters of the Steam Generators					
Number of Units	2	5.4	2	2	2
Type	Vertical U-tube with integral moisture separator	5.5	Vertical Untube with integral moisture separator	Vertical U-tube with integral with integral moisture seperator	Vertical U-tube moisture separator
Tube material	Inconel(ASME SB-163)	5.4	Inconel(ASME SB-163)	NiCrFe alloy	NiCrFe alloy
Shell material	SA-533 Gr. B Class 1 and SA-516, Gr. 70		SA-533 Gr B Class 1 and SA-516, Gr. 70	SA-533 Gr B Class 1 and SA-516, Gr. 70	SA-533 Gr. B Class 1 and SA-516, Cr. 70
Tube side design pressure, psig	2,485	5.4	2,485	2,485	2,485
Tube side design temperature, F	650	5.4	650	650	650
Tube side design flow, lb/hr	74×10^6	5.4	74×10^6	60.2×10^6	61×10^6
Shell side design pressure, psia	1,100	5.4	1,100	1,100	1,000
Shell side design temperature, F	560	5.4	560	560	550

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PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Steam Generators (Cont)					
Operating pressure, tube side, nominal, psig	2,235	5.4	2,235	2,235	2,235
Operating pressure, shell side, maxim- , psig	985		985	985	885
Maximum moisture at outlet at full load, %	0.2	5.4	0.2	0.2	0.2
Hydrostatic test pressure, tube side (cold) psig	3,110		3,110	3,110	3,110
Steam pressure, at full power, psia	900	5.4	900	900	815
Steam temperature, at full power, F	532	5.4	532	531.95	520.3
Principal Design Parameters of the Reactor Coolant Pumps					
Number of units	4	5.4	4	4	4
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge		Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge
Design pressure, psig	2,485	5.4	2,485	2,45	2,485
Design temperature, F	650	5.4	650	650	650
Operating pressure, nominal psig	2,235	5.4	2,235	2,235	2,235
Suction temperature, F	553	5.4	553	553.5	540
Design capacity, gal/min	99,000	5.4	99,000	80,000	80,000
Design head, ft	310	5.4	310	275	250
Hydrostatic test Pressure (cold), psig	3,110		3,110	3,110	3,110

TABLE 1.3-1 (Sheet 9 of 13) Revision 9 (12/97)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Reactor Coolant Pumps (Cont'd)					
Motor type	AC induction, single speed		AC induction, single speed	AC induction, single speed	AC induction, single speed
Motor rating, hp	9,700		9,700	6,500	6,500
Principal Design Parameters of the Reactor Coolant Piping					
Material	SA-516, Gr 70 with nominal 7/32 SS Clad		SA-516, Gr 70 with nominal 7/32 SS Clad	SA-516, Gr 70 with nominal 7/32 SS Clad	
Hot leg ID, in.	42	5.4	42	42	42
Cold leg ID, in.	30	5.4	30	30	30
Between pump and steam generator ID, in.	30	5.4	30	30	30
Engineered Safety Feature					
Safety injection system					
No. of high pressure pumps	3	6.3	3	3	3
No. of low pressure pumps	2	6.3	2	2	2
Containment spray					
No. of pumps	2	6.2	2	2	2
Containment fan coolers					
No. of units	4	6.2	4	4	4
Air flow capacity, each at emergency conditions, ft^3/min	35,000	6.2	31,000	50,000	55,800
Safety injection tanks, number	4	6.3	4	4	4
Emergency power Diesel-generator unit	2	8.3	4 (for two units)	2	2

TABLE 1.3-1 (Sheet 10 of 13) Revision 9 (12/97)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Reactor Coolant Piping (Cont'd)					
Containment System Parameters					
Type	Steel containment vessel with cylindrical shell, hemisperical dome and ellipsoidal bottom - ASME Code, Section III, Class MC, surrounded by reinforced concrete Shield Building		Steel-lined prestressed post tensioned con-crete cylinder, curve dome roof.	Steel-lined prestressed post tensioned con-crete cylinder, curved dome roof	Steel containment vessel with cylindrical shell, hemisperical dome and ellipsoidal bottom - ASME Code, Section III, Class B, surrounded by reinforced concrete Shield Building
Design Parameters - Containment					
Inside Diameter, ft.	140	3.8	150	116	140
Height, ft.	240.5	3.8	172	207	232
Free volume, ft^3	2,677,000	6.2	2,335,000	1,780,000	2,500,000
Reference accident Pressure, psig	44	3.8	60	54	44
Steel Thickness, in.	-		-	-	-
Vertical Wall	1.90	3.8	Not applicable	Not applicable	1.91
Hemispherical Head	0.95		Not applicable	Not applicable	0.95
Knuckles	2.25		Not applicable	Not applicable	2.25
Concrete Thickness, ft.	-		-	-	
Vertical Wall	Not Applicable	3.8	4 1/3	3 3/4	Not applicable
Dome	Not Applicable		3 3/4	3 1/4	Not applicable

TABLE 1.3-1 (Sheet 11 of 13) Revision 9 (12/97)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Reactor Coolant Piping (Cont'd)					
Design Parameters - Shield Building		3.8	Not applicable	Not applicable	
Inside Diameter, ft-	148				148
Height, ft. (top of foundation to top of dome)	249.5				230.5
Concrete Thickness, ft.					
Vertical Wall	3				3
Dome	2.5				2.5
Containment Leak Prevention and Mitigation Systems	Leak-tight penetration Automatic isolation where required.	6.2	Leak-tight pentration, and continuous steel liner. Automatic isolation where required. The exhaust from penetration rooms to vent.	Leak-tight penetration, and continuous stee liner. Automatic isolation where required.	Leak-tight penetration. Automatic isolation where required.
Gaseous Effluent Purge	Discharge through vent.	6.2	Discharge thru vent.	Discharge thru vent.	Discharge thru vent.
RADIOACTIVE WASTE MANAGEMENT SYSTEM					
Liquid Waste Processing Systems					
Reactor Coolant Waste Holdup Tank (BMS)		11.2			
Number	4		1/2	4	4
Capacity (Gal.), each	45,000		6,000/25,000	51,270	40,000

TABLE 1.3-1 (Sheet 12 of 13) Revision 11-B (06/02)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
RADIOACTIVE WASTE MANAGEMENT SYSTEM (Cont'd)					
Degasifier					
→(DRN 00-803)					
Number	Flash Tank*		1 (Gas Stripper)	1	Flash Tank
←(DRN 00-803)					
Capacity (gpm)	-				-
Concentrators					
Number	2		1 (For 2 units)	1	1
Capacity (gpm)	20		50 gpm	20	2
Gaseous Waste Processing Systems					
Waste Gas Decay Tank		11.3			
Number 3	3		6 (For 2 units)	3	3
Capacity (ft), each	600		500	300	144
Pressure (psig)	380		150	380	190
Hold-up Time (days)	60		30	30	30
ELECTRIC SYSTEMS					
Number of Offsite Circuits	7	8.2.1.1	8	3	3
Number of Incoming Lines to Startup Transformers	2	8.2	2	2	2
Number of Startup Transformers	2	8.2	4	1+1(shared)	2
Number of Main Unit Transformers (Three Phase)	2	8.2	1	3 (single phase)	2
Number of 4.16 KV Engineered Safety Features System Buses	3	8.3	3	2	3
Number of 480V Engineered Safety Features System Buses (Power Centers)	3	8.3	3	2	4
Number of 120V AC Vital Buses	8	8.3	4	4	3
Number of Standby Diesel Generators	2	8.3	2	2	2
Diesel Generator Rating (KW)	4400	8.3	4700	2850	3500

→(DRN 00-803)
(*) The Flash Tank is inactive per ER-W3-00-0225-00-00.
←(DRN 00-803)

TABLE 1.3-1 (Sheet 13 of 13) Revision 11-A (02/02)

PLANT PARAMETER COMPARISON FOR WATERFORD 3 CYCLE 1

Item	Waterford 3 Cycle 1	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
INSTRUMENTATION SYSTEMS*					
Reactor Protective System		7.2	7.2	7.2	7.2
Reactor and Reactor Coolant System		7.7.1.1 7.1.1.2	7.7.1.1 7.7.1.2	7.7.1.1 7.7.1.2	7.7.1.1 7.7.1.2
Steam and Feedwater Control System		7.7.1.3	7.7.1.3	7.7.1.3	7.7.1.3
Nuclear Instrumentation		7.2.1.1	7.2.1.1	7.2.1.1	7.2.1.1
Non-Nuclear Process Instrumentation		7.5.1.5	7.5.1.5	7.5.1.5	7.5.1.5
CEA Position Instrumentation		7.5.1.3	7.5.1.3	7.5.1.3	7.5.1.3

→ (DRN 01-758)

* This section is not suited for tabular description. SAR section numbers

have been included for the location of the detailed description of each system.

← (DRN 01-758)