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SUBJECT: Forwards marked up FSAR pages w/changes expected from adoption of Tech Specs in 890605 license amend request. Tech Spec

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JULY 1 2 1989

L-89-235

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Revised Technical Specifications
Related FSAR Revisions

On June 5, 1989, Florida Power & Light Company (FPL) submitted a proposed license amendment to replace the current technical specifications with the Revised Technical Specifications (RTS). As had been previously agreed with the NRC Staff, a listing of the updated FSAR changes that would result from adoption of the RTS was included as an attachment to the license amendment request. These FSAR revisions were scheduled to be made at the next planned submittal of the updated FSAR. We have subsequently been advised that a submittal at this time of the expected changes to the FSAR would facilitate completion of the Staff's SER. Accordingly, the attachments to this letter provide the marked up Turkey Point FSAR pages showing the expected revisions.

As discussed with the Staff, these revisions may subsequently be modified for editorial purposes or to incorporate other non-related changes as part of the FSAR update activities of 10 CFR 50.71.

Should there be any questions on this information, please contact us.

Very truly yours,

R.J. Acosta for
C. O. Woody
Acting Senior Vice President - Nuclear

COW/TCG/gp

Attachment

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

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ATTACHMENT I

FSAR CHANGES DUE TO T/S CHANGES

TS No.	Title	FSAR Page
4.1.2.1.a	Heat Tracing & Boron Conc.	9.2-6
3.1.2.4.a.2	Heat Tracing & Boron Conc.	9.2-6a
3.1.2.5.a.2	Heat Tracing & Boron Conc.	9.2-23
3.4.9.2	Pressurizer P/T Limits and Pressurizer-Max Spray Temp Diff.	4.2-18 (Rev. 7)
3.4.9.3	PORV Lift Press	T 4.1-1
3.5.1.b	Accumulator Volume	T 6.2-4
3.3.2	R-11 and R-12 Setpoints	T 11.2-7a (Rev.7)
3.6.1.4/	Cont. Pressure and Temperature	7.2-36
3.6.1.5		5.1.1-1 (Rev. 7)
3.6.6	Post Accident Vent System Limits	9.12-2
3.7.1.1	Main Steam Safety Lift Settings	10.3-3
3.9.10/	RV & SFP Water Levels	11.2-8
3.9.11		14E**
5.2.1	Cont Parameters	5.1.2-1
5.6.1.1.d&e	U-235 in Fuel Racks	14E**
T 5.7-1	Cycle Limits	T 4.1-8 (Rev. 7)
4.6.1.6.a&b	Cont Structural Integrity Tendons	5.1.7-5
		5.1.7-6
3.9.1	Boron Conc. RCS/Refueling	1.3-26 (Rev. 7)
		6.2-16
		3.2.1-2 (Rev. 7)
		14.1.5-3 (Rev.7)
		14.2.1-3
		14A-6
		14B-6 (Rev. 7)
		9.5-1
		9.5-6
		9.5-16
5.6.1.2	Fuel Storage Criticality	T 9.5-1
3.9.14, 5.6	T/S Reference Numbers	14D-7
		14D-54
3.1.1.3	MTC	14.1.10 (Rev.7)*
		14.B (Rev. 7)*
3.4.2.1&2	Pressurizer Safety Lift Settings	T4.1-1
		T 4.1-3
T 2.2-1	ESF and Reactor Protection System Setpoints	7.2-1
T 3.3-3		

*These sections have been completely rewritten per Rev. 7 The entire section is attached.

**Section 14E of the FSAR will be deleted when the SFP rerack is complete. (Estimated completion: mid-1989)

ATTACHMENT II

MARKED-UP FSAR PAGES

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen within this tank is supplied to the reactor coolant for maintaining a low oxygen concentration. Fission gases are periodically removed from the system by venting the volume control tank to the Waste Disposal System.

The charging pumps take suction from the volume control tank and return the coolant to the Reactor Coolant System through the tube side of the regenerative heat exchanger.

The cation bed demineralizer, located downstream of the mixed bed demineralizers, is used intermittently to control cesium activity in the coolant and also to remove excess lithium which is formed from B^{10} (n, α) Li^{7} reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of approximately 12 percent by weight. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low pressure steam. A transfer pump is used to transfer the batch to the boric acid tanks. Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with primary water as makeup for normal leakage or for increasing the reactor coolant boron concentration during normal operation. Electric immersion heaters maintain the temperature of the boric acid tanks solution high enough to prevent precipitation.

Electrical heat tracing is provided in conjunction with insulation on all piping, line mounted instrumentation and components normally containing concentrated boric acid solution. All such piping requiring this heat tracing is located in the auxiliary building. The heat tracing is designed to maintain the temperature of the piping and contents at 160 F to 180 F with an ambient air temperature of 40 F. In the event the building temperature should fall an additional 20 F the contents of the piping would be maintained at least at 140 F. This is ~~well~~ above ~~120 F,~~ the temperature that nominal 12 percent boric acid solution begins to precipitate.

Based on one heat tracing train in operation for each component, to assure continuous solution temperatures above the precipitation temperature ~~130°F~~; temperature indications over 145° F were specified because of the complex geometry generated by the tracing installation and by the limited number of measurement monitors.

Boric Acid Tanks

The boric acid tank capacities are sized to store sufficient boric acid solution for refueling plus enough boric acid solution for a cold shutdown shortly after initial full power operation is achieved. In addition, each tank has sufficient boric acid solution to achieve cold shutdown if the most reactive RCCA is not inserted. One tank is normally used with each unit and a third tank serves as a shared standby.

The concentration of boric acid solution in storage is maintained ~~between~~ at a nominal 12% ~~11.5 and 12.5%~~ by weight. Periodic manual sampling is performed and corrective action is taken, if necessary, to ensure that these limits are maintained. Therefore, measured amounts of boric acid solution can be delivered to the reactor coolant to control the concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tanks are constructed of austenitic stainless steel.

Boric Acid Tank Heaters

Two 100% capacity electric immersion heaters located near the bottom of each boric acid tank are designed to maintain the temperature of the boric acid solution at 165 F with an ambient air temperature of 40 F thus ensuring a temperature in excess of the solubility limit. ~~(for 20,000 ppm boron this is 130 F).~~ The temperature is monitored and low temperature is alarmed in the control room. The heaters are sheathed in austenitic stainless steel.

Batching Tank

The batching tank is sized to hold one week's makeup supply of boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of 1/2 gpm at beginning of core life. The tank may also be used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank or for draining the tank.

The methods used to measure the initial NDTT of the reactor vessel base plate material are given in Appendix 4A.

4.2.6 MAXIMUM HEATING AND COOLING RATES

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.5. During unit heatup and cooldown, the rates of temperature and pressure changes are limited. The system design heatup and cooldown rate of 100°F per hour satisfies stress limits for cyclic operation (ASME B&PV Code, Section III) and is consistent with the expected number of cycles. However, the normal system heatup and cooldown rate is conservatively set at 50°F per hour. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F per hour. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant. The allowable cooldown rate for the pressurizer is 200°F per hour.

The allowable heatup ~~and cooldown~~ rate for the pressurizer is ~~200~~ 100°F per hour. ~~Because of its thinner wall sections.~~ The stresses are within acceptable limits for the anticipated usage. A maximum temperature difference (ΔT) of 320°F between the pressurizer and reactor coolant system is specified up to a maximum pressurizer temperature of 500°F (References 1 and 2). This allows steam bubble formation at an earlier time during startup to reduce the chances of an overpressure event by reducing the period during which the plant is solid. At pressurizer temperature greater than 500°, ΔT is specified as 200°F with a minimum of 100°F. Spray actuation transients during the condition of ΔT greater than 100°F shall be limited to those in Table 2-2, Figure 2-1 and Figure 2-5 in Reference 1.

The fastest cooldown rates which result from the hypothetical case of a break of a main steam line are discussed in Section 14.

4.2.7 LEAKAGE

The existence of leakage from the Reactor Coolant System to the containment regardless of the source of leakage, is detected by one or more of the following conditions:

TABLE 4.1-1

REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

Total Primary Heat Output, MWt	2208
Total Primary Heat Output, Btu/hr	7535×10^6
Number of Loops	3
Coolant Volume (liquid), including total pressurizer volume, ft^3	9343
Total Reactor Coolant Flow, gpm	265,500

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485±1%
Power Relief Valves	
i) Normal Operation	2335
ii) OMS Actuation During Heatup and Cooldown	
a) RCS ≤ 285°F	415±15
	Setpoint increases step-wise:
b) RCS 319°F	495
RCS 347°F	600
RCS 384°F	832.5
RCS 421°F	1147.5
RCS 472°F	1710
RCS 508°F	2220
RCS 554°F	2335
RCS 750°F	2335
Pressurizer Spray Valves (Open)	2260
High Pressure Trip	2385
High Pressure Alarm	2310
Low Pressure Trip	1835
Low Pressure Alarm	2185
Hydrostatic Test Pressure	3107

TABLE 6.2-4

ACCUMULATOR DESIGN PARAMETERS

Number	3
Type	Stainless steel clad/ carbon steel
Design pressure, psig	700
Design temperature, F	300
Operating temperature, F	70 - 120
Normal pressure, psig	660
Minimum pressure, psig	600
Total volume, ft^3	1200
Minimum water volume at operating conditions, ft^3	~~775~~ 875
Boron concentration, ppm	1950 +400, -0 ppm
Relief valve set point, psig*	700

* The relief valves have soft seats and are designed and tested to ensure zero leakage at normal operating pressure

TABLE 11.2-7

RADIATION MONITORING SYSTEM CHANNEL SENSITIVITIES

Channel	Sensitivity Range	Detected Isotopes
Process		
R3-11 & R4-11	1.0×10^{-9} to 1.0×10^{-6}*	I^{131},I^{133},Cs^{134},Cs^{137}
R3-12 & R4-12	1.0×10^{-6} to 1.0×10^{-3}*	Kr^{85},Ar^{41},Xe^{135},Xe^{133}
R-14	5.0×10^{-7} to 1.0×10^{-4}*	Kr^{85},Ar^{41},Xe^{135},Xe^{133}
R3-15 & R4-15	1.0×10^{-6} to 1.0×10^{-3}*	Kr^{85},Ar^{41},Xe^{135},Xe^{133}
R3-17A,R3-17B, R4-17A, R4-17B	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60},Mixed Fission Products
R-18	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60},Mixed Fission Products
R3-19, R4-19	1.0×10^{-5} to 1.0×10^{-2}*	Co^{60},Mixed Fission Products
R3-20, R4-20	1.0×10^{-1} to $1.0 \times 10^{+5}$**	Kr^{85},Ar^{41},Xe^{133},Xe^{135}
Area R1 thru R24	1.0×10^{-1} to $1.0 \times 10^{+3}$**	

Note: Prefixes R3 or R4 designates Unit #3 or Unit #4. Channels without prefix number monitor both units.

TABLE 11.2-7a

RADIATION MONITORING SYSTEM CHANNEL ALARM SET POINTS

Channel	Alarm Set Point***	Basis
Process		
R3-11 & R4-11	5.4×10^{-7}*	Containment Purge = ~~70,000 cfm~~ 35,000 cfm/unit 0-30 Day X/Q=4.32 x 10^{-6} sec/m^3
R3-12 & R4-12	7.9×10^{-5}*	Containment Purge = ~~70,000 cfm~~ 35,000 cfm/unit 0-12 Day X/Q=1.15 x 10^{-4} sec/m^3
R-14	5.0×10^{-5}*	Vent Air Flow=110,000 cfm 0-12 Day X/Q=1.15 x 10^{-4} sec/m^3
R3-15 & R4-15	2.7×10^{-4}*	Blowdown Rate=96,000 lbs/hr Circulating Water Flow=157,000 gpm
R3-17A,R3-17B R4-17A,R4-17B	5.1×10^{-4}*	Coolant Surge Volume=1,000 gal 0-2 hour X/Q=1.5 x 10^{-4} sec/m^3
R-18	1.3×10^{-4}*	Water Effluent flow=20 gpm Circulating Water Rate=157,000 gpm
R3-19 & R4-19	3.7×10^{-5}*	Blowdown Rate=96,000 lbs/hr Circulating Water Flow=157,000 gpm
R3-20 & R4-20	1.5 x previous week's average or $1.0 \times 10^{\circ}$** whichever is greater	Letdown Rate=120 gpm

Notes: * is given in μCi /cc
** is given in mr/hr
*** above instrument background count

Rev 6 7/88

0081F/0029F

Steam Line Pressure

Three pressure channels per steam line are used for steam line break protection (low steam line pressure in two out of three steam lines with high steam flow in two out of three lines actuates safety injection).

Normal Operating Environment

The control room is maintained at the personnel comfort level of (70 ± 10) F. Protective equipment inside the room is designed to operate within design tolerance over this temperature range and will perform its protective function in an ambient of 120F and 95% relative humidity (i.e., there will be no loss-of-function in an ambient temperature of 120F°).

The operating environment for equipment within the containment will normally be controlled to less than 120 F. The Reactor Protective System instrumentation within the containment is designed for continuous operation. The temperature of the out-of-core neutron detectors is maintained at or below 135 F by the normal containment air cooling system. The detectors are designed for continuous operation at 135 F and will withstand operation at 175 F for short durations.

Typical test data (or reasonable engineering extrapolation based on test data) will be used to verify that protection systems equipment will meet, on a continuing basis, the functional requirements under the anticipated normal ambient conditions.

The average air temperature may be between 120 and 125°F for 336 equivalent hours during a calendar year.

5 STRUCTURES

5.1 CONTAINMENT STRUCTURE

5.1.1 DESIGN BASIS

The containment structure completely encloses the reactor coolant system to minimize release of radioactive material to the environment should a failure of the coolant system occur. The structure provides adequate biological shielding for both normal operation and the hypothetical accident condition.

The containment structure is designed to withstand a pressure of 59 psig and 283°F. The original transient analysis calculated peak accident pressure is 49.9 psig and the peak accident temperature is 276°F. The higher design pressure and temperature is based on the PSAR commitment which at that time did not take any credit for the accumulators. The subsequent transient analysis yielded the lower pressure and temperature; however, for a check in the structural integrity of the containment, 55 psig is considered as nominal structural design pressure, thus allowing a margin of 10% over the calculated peak accident pressure.

The principal design basis for the structure is that it should be capable of withstanding, without loss of integrity, the peak pressure resulting from any size pipe break including the maximum hypothetical accident (MHA). The MHA is defined as the release of the water in the system through a double-ended break of a reactor coolant pipe, coincident with a loss of normal power. The subsequent pressure behavior is determined by the engineered safeguards and the combined influence of energy sources and heat sinks as described in Section 14.3.4.

permit any equipment repair in case of malfunction either before or following initial operation.

Containment Air Supply

Two independent full-capacity air supply lines are provided to missile-protected areas in each containment, one entering through the Service Air penetrations (#34). The alternative air supply could be fed through the containment Air Sample Return penetrations (#32). Both supplies are capable of sustaining the design flow of 40 scfm.

Containment Collection Headers and Exhaust

Two independent full-capacity collection headers, each consisting of a two inch line with 90 - 1/4 inch holes on six inch centers, are mounted approximately 120 degrees apart on radii of the containment dome liner. These lines are permanently run through missile-protected areas and connected to separate penetrations (#16 and #53), which feed separate manual L.C. double-valves and then are connected together outside containment. The common header goes, in turn, through a flow meter orifice, past a filter test and condensate drain connection, through HEPA and impregnated charcoal filters, past filter outlet temperature, sample and filter test connections to the vent collection header for the gas handling equipment. A differential pressure gauge is connected across the series filters. An alternate direct flow path is provided to the suction of the Auxiliary Building Exhaust Fans.

9.12.1.3 Components

Filters	Number	Air Flow	Des. Temp.	Differ. Pres. Clean	Differ. Pres. Loaded
Absolute (HEPA)	1	55 cfm	180F	1" W.G.	3~~2~~" W.G.
Charcoal, Iodine Impregnated	1	55 cfm	180F	1" W.G.	3~~2~~" W.G.

10.3.2 SECONDARY-PRIMARY INTERACTIONS

Following a turbine trip, the control system reduces reactor power output immediately by a reactor trip.

In the event of failure of one feedwater pump the feedwater pump remaining in service will carry approximately 60 percent of full load feedwater flow. If both normal feedwater pumps fail, the turbine will be tripped, and the emergency feedwater pumps will start automatically.

10.3.3 PRESSURE RELIEF

Pressure relief is required at the system design pressure of 1085 psig, and the first safety valves is set to relieve at this pressure. Additional safety valves are set at pressures up to 113~~9~~0 psig. ~~as allowed by the ASME B&PV Code, Section VIII.~~ The code allows a 1% tolerance on the safety valve setpoint. Manual means are provided for operating the 10% steam dump to atmosphere. The SDTA valves are designed to operate in the unlikely event of complete loss of electrical power and/or instrument air .

The pressure relief capacity is equal to the steam generation rate of maximum calculated conditions.

10.3.4 SYSTEM INCIDENT POTENTIAL

The evaluation of the capability to isolate a steam generator to limit the loss of radioactivity is presented in Section 14.2.4. The steam line break accident analysis is presented in detail in Section 14.2.5

Fuel Handling Shield

The refueling cavity is irregularly shaped, formed by the upper portions of the primary shield concrete, and other sidewalls of varying thicknesses. A portion of the cavity is used for storing the upper and lower internals packages. The walls vary in thickness, from 4 to 5 ft.

The refueling cavity, flooded with borated water to elevation 56'-10" ~~57'-4"~~ during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 23 ~~25~~ ft. above the reactor vessel flange. This height ensures that a minimum of 9 ft. of water will be above the active fuel of a withdrawn fuel assembly. Under these conditions, the dose rate from the active fuel is less than 2.5 mr/hr at the water surface.

The spent fuel assemblies and RCC assemblies are remotely removed from the containment through the horizontal spent fuel transfer tube to be placed in the spent fuel pit. Concrete, 3' to 4'6" thick, shields the spent fuel transfer tube. This shielding is designed to protect personnel from radiation during a time a spent fuel assembly is passing through the main concrete support of the containment and the transfer tube.

Radial shielding as the spent fuel is raised for transfer to the spent fuel storage pit is provided by the water and concrete walls of the fuel transfer canal. Administrative procedures ensure that no personnel will receive more than 15 mr/hr above ambient background in the area adjacent to the spent fuel transfer pit.

Fuel is stored in the spent fuel pit of the Auxiliary Building which is located adjacent to the containment. Shielding for the spent fuel storage pit is provided by 5' 6" feet thick concrete walls to elevation 32' 10"; above this elevation the walls are tapered in places to a thickness of 3 ft. The pit is flooded to a level such that the water height is 23 ~~24~~ feet ~~6 inches~~ above the stored spent assemblies. During spent fuel handling a minimum of 7 feet 11 inches is maintained above the top of a fuel assembly.

5.1.2 GENERAL DESCRIPTION AND DESIGN LOADS

The containment, which is a Class I structure, consists of a post-tensioned reinforced concrete cylinder and a shallow dome, connected to and supported by a massive reinforced concrete foundation slab as shown in Figure 5.1-1.

The inside surface of the structure is lined with a ¼" thick welded steel plate to insure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the containment through welded steel penetrations as shown in Figure 5.1-2 and 5.1-3. These penetrations and all other areas of the liner plate not backed by structural concrete are designed, fabricated, inspected, and installed in accordance with Section III, Subsection B, of the ASME Pressure Vessel Code.

Principal dimensions of the containment structure are as follows:

Inside diameter	116 feet
Inside height (including dome)*	~~169~~ 170.6 feet
Vertical wall thickness	3 3/4 feet
Dome thickness	3 1/4 feet
Foundation slab thickness	10 1/2 feet
Internal free volume	1,550,000 cu. ft.

In the concept of a post-tensioned containment, the internal pressure load is balanced by the application of an opposing external pressure type load on the structure. Sufficient post-tensioning is applied on the cylinder and

* The inside height does not include a nominal 1.6 ft concrete pad on top of the baseplate. Actual inside height including baseplate is 169 ft.

TABLE 4.1-8

DESIGN THERMAL AND LOADING CYCLES - 40 YEARS

Transient Design Condition	Design Cycles	Expected Cycles
1. Station heatup at 100°F per hour	200 (5/yr)	80
2. Station cooldown at 100°F per hour	200 (5/yr)	80
3. Station loading at 5% of full power/min	14,500 (1/day)	2500
4. Station unloading at 5% of full power/min	14,500 (1/day)	2500
5. Step load increase of 10% of full power (but not to exceed full power)	2,000 (1/week)	500
6. Step load decrease of 10% of full power	2,000 (1/week)	500
7. Step load decrease of 50% of full power	200 (5/year)	20
8. Reactor trip	400 (10/year)	40
9. Hydrostatic test at 3107 psig pressure, 100°F temperature	5 (pre-operational)	2
10. Hydrostatic test at 2435 psig pressure and 400°F temperature	150 (post-operational)	30

11. Steady state fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur.

5.1.7.4 Tendon Surveillance

Provisions are made for an in-service tendon surveillance program, throughout the life of the plant that will maintain confidence in the integrity of the containment structure. This program is supplemented by a corrosion control program.

The following quantity of tendons have been provided over and above the structural requirements, and are available for inspection and lift-off readings:

Horizontal -	Five tendons ~~Three 120 degree tendons comprising one complete hoop system.~~
Vertical -	Four ~~Three~~ tendons ~~spaced approximately 120 degrees apart.~~
Dome -	Three tendons ~~spaced approximately 120 degrees apart.~~

The tendons chosen for surveillance are a random but representative sample.

The surveillance program for structural integrity and corrosion protection consists of the following operations to be performed during each inspection:

(a) Lift-off readings will be taken for all of the ~~nine~~ twelve tendons.

(b) One tendon of each directional group will be relaxed and one wire from each relaxed tendon will be removed as samples for inspection. Since these tendons are in excess of those required by design, these samples need not be replaced.

(c) After the inspection, the tendons will be retensioned to the stress level measured at the lift-off reading and then checked by a final lift-off reading.

(d) Should the inspection of one of the wires reveal any significant corrosion (pitting, or loss of area), further inspection of the other two sets will be made to determine the extent of the corrosion and its significance to the load-carrying capacity of the structure. Samples of corroded wire will be tested to failure to evaluate the effects of any corrosion on the tensile strength of the wire.

The inspection of the ~~three~~ four vertical tendons ~~spaced 120 degrees apart~~ in the wall is sufficient to indicate any tendon corrosion that could possibly appear longitudinally along the full height of the structure. Furthermore, the vertical tendons extend below the ground water table where corrosion is most likely to occur, if at all. Therefore, the ~~nine~~ twelve tendons arranged as described will provide adequate corrosion surveillance.

Reference Sections:

Section Title	Section
Containment	5.1
Engineered Safety Features	6
Electrical System	8.1, 8.2

1.3.8 FUEL AND WASTE STORAGE SYSTEMS (GDC 66-GDC 69)

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than prescribed locations. Borated water is used to fill the spent fuel storage pit at a concentration to match that used in the refueling cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.95$. Criticality of the fuel assemblies in the spent fuel rack is prevented by the inherent design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between the assemblies.

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shutdown the core to a $k_{eff} = 0.95$. This shutdown margin maintains the core at $k_{eff} < 0.99$, even if all control rods are withdrawn from the core. Periodic checks of refueling water boron concentration ensure the proper shutdown margin.

The design of the fuel handling equipment incorporates built-in interlocks and safety features, the use of detailed refueling instructions and observance of minimum operating conditions provide assurance that no incident could occur during the refueling operations that would result in a risk to public health and safety.

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal is accomplished with an auxiliary cooling heat exchanger.

The water in the tank is borated to a concentration which assures reactor shutdown by at least ~~10~~ 5% $\Delta k/k$ when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.4 weight percent boric acid. At 32°F the solubility limit of boric acid is 2.2%. Therefore the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F.

A technical specification minimum level alarm and a high level alarm are provided. Nominal RWST level indication with high and low level alarms are also provided. The low level alarm setpoint has been determined to provide sufficient NPSH for the containment spray pumps.

A dynamic response analysis similar to that performed for the Containment Structure has been performed to determine the horizontal loads applied to this tank for 5% ground acceleration based on yield stresses and a 15% ground acceleration based on maximum deflection. Waves generated in the tank have been taken into account as per "Nuclear Reactors and Earthquake", TID 7024. A membrane stress analysis of the vertical cylindrical tank was performed considering the discontinuities at the base and top.

The design parameters are given in Table 6.2-6.

Safety Injection Pumps

The four high-head safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electrical motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. The design parameters are presented in Table 6.2-7 and Figure 6.2-3 gives the performance characteristic of these pumps.

The two residual heat removal (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure to the Reactor

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling has been established for Cycle 1 as shown in Table 3.2.1-1, line 29. This concentration together with the RCCA provides approximately 10 percent shutdown for these operations. In Reference 17 of Section 3.2.1, the refueling shutdown margin has been revised to 5 percent (delta k)/k. The concentration is also sufficient to maintain the core shutdown without any RCCA during refueling. For cold shutdown, at the beginning of core life, a concentration (shown in Table 3.2.1-1, line 37) is sufficient for one percent shutdown with all but one stuck rod inserted. The boron concentration (Table 3.2.1-1, line 29) for refueling is equivalent to less than two percent by weight boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial full power boron concentration without equilibrium xenon and samarium is specified in Table 3.2.1-1, line 34. As these fission product poisons are built up, the boron concentration is reduced to that specified in Table 3.2.1-1, line 36.

This initial boron concentration is that which permits the withdrawal of the control banks to their operational limits. The xenon-free hot, zero power shutdown (k = 0.99) with all but one stuck rod inserted, can be maintained with the boron concentration specified in Table 3.2.1-1, line 38. This concentration is less than the full power operating value with equilibrium xenon.

c) The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution,

d) The minimum boron concentration of the reactor coolant system is 1950 ppm, corresponding to a shutdown of at least 5 percent delta k/k with all control rods in; periodic sampling ensures that this concentration is maintained, and

e) Neutron sources installed in the core increase subcritical multiplication to a level that can be detected by out-of-core BF_3 detectors. Instrumentation associated with these detectors provide an audible signal of core flux count.

A minimum water volume in the Reactor Coolant System of 3200 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop. The maximum dilution flow of 231 gpm and uniform mixing are also considered.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count rate increase is proportional to the inverse multiplication factor.

The boron concentration must be reduced from greater than 1950 ppm to approximately 1450 ppm before the reactor will go critical. This would take at least 30.2 minutes. This is ample time for the operator to recognize high count rate signal and isolate the primary water makeup source by closing valves.

Only one fuel assembly can be handled at a time.

Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

Administrative control will be used to prevent the handling of heavy objects, such as a spent fuel shipping container, above the fuel racks.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit.

Should a spent fuel assembly become stuck in the transfer tube, natural convection will maintain adequate cooling. The fuel handling equipment is described in detail in Section 9.5.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and a bell and light in the control room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least ~~10~~ 5 per cent with all rod cluster control assemblies inserted. At this boron concentration the core would also be ~~more than 2 per cent~~ subcritical with all control rods withdrawn. The refueling cavity is filled with water meeting the same boric acid specification.

All these safety features make the probability of a fuel handling incident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

2.2 Nuclear Design

The Cycle 11 loading pattern results in a maximum F_Q less than 2.32 at normal operating conditions. In Table 2, a comparison is provided of the range of values encompassing the Cycle 11 core kinetics parameters with the current limit based on previously submitted accident analyses. It can be seen from the table that the Cycle 11 range of values fall within the current limits. These parameters are evaluated in Section 3.0. In Table 3, the control rod worths and requirements are provided. The required shutdown margin is based on previously submitted accident analyses.[7] The reactivity defects encompass the values for the Cycle 11 core. The available shutdown margin meets or exceeds the minimum required.

~~The refueling boron concentration required to maintain at least 10 percent shutdown margin for Cycle 11 is greater than 1950 ppm.~~ The minimum refueling boron concentration requirement is ~~therefore~~ set at ~~2250~~ 1950 ppm which maintains at least ~~10%~~ 5% Δk/k shutdown margin. The decreased shutdown margin from previous cycles ~~increase~~ is due to the large amount of excess reactivity installed at BOC for the long cycle length. ~~Additionally, 2250~~ 1950 ppm soluble boron concentration is sufficiently high to prevent criticality with all rods out during refueling. The boron dilution event has been evaluated and the conclusions presented in the FSAR are still valid.

The loading contains three different types of burnable absorbers which are described and distributed as follows: 8 Region 13A fuel assemblies contain a total of 64 new WABA rods[13], 16 Region 13C assemblies contain 1408 new IFBA[4] fuel rods, 8 Region 13C assemblies contain 480 new IFBA fuel rods and 12 Region 11A fuel assemblies contain a total of 240 reduced-length hafnium burnable absorber rods. The hafnium burnable absorber rods are being reinserted from Cycle 10. The hafnium rods are modeled in the Cycle 11 core by approved Westinghouse design codes, as was done for the Unit 3 Cycle 10 core [14]. Burnable absorber lengths and the locations of the source rods and burnable absorber rods are shown in Figure 2.

9.5 FUEL HANDLING SYSTEM

The Fuel Handling System provides a safe effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. The system is designed to minimize the possibility of malfunction that causes fuel damage and potential fission product release.

The Fuel Handling System consists basically of the refueling cavity, the spent fuel pit, and the Fuel Transfer System.

9.5.1 DESIGN BASIS

Prevention of Fuel Storage Criticality

Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. (GDC 66)

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The new and spent fuel storage pits have accommodations as defined in Table 9.5-1. In addition, the spent fuel pit has an area set aside for accepting the spent fuel shipping casks. Cask handling is also done under water. Borated water is used to fill the spent fuel storage pit at a concentration to match that used in the refueling cavity and refueling canal during refueling operations. The fuel in the spent fuel and new fuel storage pits is stored vertically in an array with the sufficient center-to-center distance between assemblies to assure Keff $\leq$ 0.95 for the spent fuel storage pit and Keff $\leq$ 0.~~90~~ 95 for the new fuel storage pit.

by two concrete shielding walls, which extend upward to the same elevation as the refueling cavity. The floor of the canal is at a lower elevation than the refueling cavity to provide the greater depth required for the fuel transfer system tipping device and the control cluster changing fixture located in the canal. The transfer tube enters the reactor containment and protrudes through the end of the canal. The canal walls and floor are lined with stainless steel.

Refueling Water Storage Tank

The normal duty of the refueling water storage tank is to supply borated water to the refueling canal for refueling operations. In addition, the tank provides borated water for delivery to the core following either a loss-of-coolant or a steam line rupture accident. This is described in Chapter 6.

The capacity of the tank is based upon the requirement for filling the refueling cavity and refueling canal.

The water in the tank is borated to a concentration which assures reactor shutdown by at least ~~10~~ 5% $\delta k/k$ when all RCC assemblies are inserted and the reactor is cooled down for refueling.

The tank design parameters are given in Chapter 6.

Spent Fuel Storage Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor.

The pit design parameters are listed in Table 9.5-1. Control rods are stored in fuel assemblies.

Spent fuel assemblies are handled by a long handled tool suspended from the spent fuel pit bridge overhead crane and manipulated by an operator standing on the movable bridge over the pit.

9.5.3 SYSTEM EVALUATION

Underwater transfer of spent fuel provides essential ease and corresponding safety in handling operations. Water is an effective, economic and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

a) Gamma radiation levels in the containment, control room and fuel storage areas are continuously monitored (see Section 11.2.3). These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the control room of an abnormal core flux level.

b) Containment integrity is maintained when the reactor vessel head is removed unless the shutdown margin is maintained greater than 5 ~~10~~% $\Delta k/k$.

c) Whenever any fuel is being added to the reactor core or is being relocated, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

<u>Incident Protection</u>

Direct communication between the control room and the refueling cavity manipulator crane is available whenever changes in core geometry are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe conditions detected from the control board indicators during fuel movement.

<u>Malfunction Analysis</u>

An analysis is presented in Section 14 concerning damage to one complete outer row of fuel elements in an assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel handling incident.

TABLE 9.5-1
FUEL HANDLING DATA

New Fuel Storage Area	
Core storage capacity	1/3
Equivalent fuel assemblies	53
Center-to-center spacing of assemblies, in.	21
Maximum K_{eff}, if flooded with unborated water	~~0.90~~ 0.95
Spent Fuel Storage Pit Unit 4	
Core storage capacity	Approx 4
Equivalent fuel assemblies	621
Number of space accomodations for spent fuel shipping casks	1
Center-to-center spacing of assemblies, in.	13.659
Maximum K_{eff}, if flooded with unborated water	<0.95
Spent Fuel Storage Pit Unit 3	
Core storage capacity	Approx 9
Equivalent fuel assemblies	1404
Number of space accomodations for spent fuel shipping casks	1
Center-to-center spacing of assembiles, in.	
Region 1	10.6
Region 2	9.0
Maximum K_{eff}, if flooded with unborated water	<0.95
Miscellaneous Details	
Width of refueling canal, ft	3
Wall thickness of spent fuel storage pit, ft	3 to 6
Weight of fuel assembly with RCC (dry), lb.	≈1580
Capacity of refueling water storage tank, gal.	338,000
Minimum contents of refueling water storage tank for Safety Injection or Spray System Operability, gal.	320,000
Quantity of water required for refueling, gal	285,000
Minimum water required for post MHA sump recirculation pump N.P.S.H. protection including allowance for possibility of drainage delay in containment	249,000

[illegible]

density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not excessively deformed and the dropped assembly has more than eight inches of water separating it from the active fuel height of stored assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity. These would include the inadvertant drop of an assembly between the outside periphery of the rack and the pool wall, a cask drop accident, or damage to the fuel racks when empty rack modules are being installed. Therefore, for accident conditions, the double contingency principle of ANSI N16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to provide for protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water, as specified in ~~proposed~~ Technical Specification ~~3.17~~ 3.9.14, is a realistic initial condition.

The presence of approximately 1950 ppm boron in the pool water will decrease reactivity by about 30 percent Δk. In perspective, this is more negative reactivity than is present in the poison plates (25 percent Δk), so k_{eff} for the rack would be less than 0.95 even if the poison plates were not present. Thus, for postulated accidents, should there be a reactivity increase, k_{eff} would still be less than or equal to 0.95 due to the combined effects of the dissolved boron and the poison plates.

The "optimum moderation" accident is not a problem in spent fuel storage racks because the presence of poison plates removes the conditions necessary for "optimum moderation". The k_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 in the poison rack design.

3.1.3 Calculation Methods

3.1.3.1 Criticality Analysis for Region I

The calculation method and cross-section values for Region I are verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. These benchmarking data are sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which provides for the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes [2,3] for cross-section generation and KENO IV[4] for reactivity determination.

4.7.3 Quality Assurance

The design, procurement, and fabrication of the new high density spent fuel storage racks comply with the pertinent Quality Assurance requirements of Appendix B to 10 CFR 50 as implemented through FPL's Topical Quality Assurance Report FPL-NQA-100A[9]; the Westinghouse Water Reactors Division Quality Assurance Plan as described in WCAP 8370[10]; and the Bechtel Quality Assurance Program for Nuclear Plants, BQ-TOP-1[11], all approved by the NRC.

4.7.4 Construction Techniques

4.7.4.1 Administrative Controls During Manufacturing and Installation

The Turkey Point Units 3 and 4 new spent fuel storage racks will be manufactured at the Westinghouse Nuclear Components Division, Pensacola, Florida. This facility is a modern high-quality shop with extensive experience in forming, machining, welding, and assembling nuclear-grade equipment. Forming and welding equipment are specifically designed for fuel rack fabrication and all welders are qualified in accordance with ASME Code Section IX.

To avoid damage to the stored spent fuel during rack replacement, all work on the racks in the spent fuel pool area will be performed by written procedures. These procedures preclude the movement of the fuel racks over the stored spent fuel assemblies.

Radiation exposures during the removal of the old racks from the pool will be controlled by written procedures. Water levels will be maintained to afford adequate shielding from the direct radiation of the spent fuel. Prior to rack replacement, the cleanup system will be operated to reduce the activity of the pool water to as low a level as can be practically achieved.

39.14 5.6

During rack installation, it will be necessary to temporarily store some Region I fuel assemblies in the Region II spent fuel racks. ~~Proposed~~ Technical Specifications ~~3.17~~ and ~~5.4~~ (see Section 3.5) describe the administrative controls that will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks.

4.7.4.2 Procedure

4.7.4.2.1 Preinstallation

The following sequence of preinstallation events is anticipated for the spent fuel storage rack replacement for Units 3 and 4:

a. Design and fabricate new spent fuel storage racks.

b. Prepare modification procedure.

evaluated for a complete loss of load from full power without a direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The Reactor Coolant System and Steam System pressure relieving capacities are designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or turbine bypass control systems.

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 102 percent of full power, without direct reactor trip, primarily to show the adequacy of the pressure-relieving devices, and also to demonstrate core protection margins; i.e., the turbine is assumed to trip without actuating all the sensors for reactor trip on the turbine stop valves. This assumption delays reactor trip until conditions in the reactor coolant system (RCS) result in a trip due to other signals. Thus, the analysis assumes a worst transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program LOFTRAN[1]. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables, including temperatures, pressures, and power level.

The major assumptions used in the analysis are summarized below:

A. Initial Operating Conditions

The initial reactor power and RCS temperatures are assumed at their maximum values consistent with steady-state, full-power operation, including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value consistent with steady-state, full-power operation, including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss and the minimum margin to core protection limits at the initiation of the accident.

B. Reactivity Coefficients

Two cases are analyzed:

1. Minimum Reactivity Feedback

   A least negative moderator temperature coefficient and a least negative Doppler-only power coefficient are assumed.

2. Maximum Reactivity Feedback

   A conservatively large negative moderator temperature coefficient and a most negative Doppler-only power coefficient are assumed.

C. Reactor Control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. Steam Release

No credit is taken for the operation of the turbine bypass system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

E. Pressurizer Spray and Power-Operated Relief Valves

Two cases for both the minimum and maximum reactivity feedback are analyzed:

1. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

2. No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

F. Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow, since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

G. Reactor Trip

Reactor trip is actuated by the first reactor protection system trip setpoint reached, with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, and low-low steam generator water level.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. Normal RCS and engineered safety systems are not required to function. However, cases are analyzed both with and without the operation of pressurizer spray and power operated relief valves to ensure that the worst case is presented.

RESULTS

The transient responses for a total loss of load from 102 percent of full-power operation are shown for four cases: Two cases for minimum reactivity feedback and two cases for maximum reactivity feedback (Figures 14.1.10-1 through 14.1.10-8). The calculated sequence of events for the accident is shown in Table 14.1.10-1.

Figures 14.1.10-1 and 14.1.10-2 show the transient responses for the total loss of steam load with minimum reactivity feedback, assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam bypass. The reactor is tripped by high pressurizer pressure trip channel. The minimum departure from nucleate boiling ratio (DNBR) remains well above the limit value. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint. The pressurizer pressure raises to a maximum of 2509 psia before decreasing.

Figures 14.1.10-3 and 14.1.10-4 show the responses for the total loss of steam load with maximum reactivity feedback. All other plant parameters are the same as the above. The DNBR increases throughout the transient and does not drop below its initial value. The pressurizer safety valves and steam generator safety valves prevent overpressurization in the primary and secondary systems. The pressurizer pressure increases to 2355 psia initially. The rise in the RCS average temperature of approximately 16.5°F causes a reduction in neutron flux due to reactivity feedback effects, resulting in a decrease in pressurizer pressure.

The loss of load accident was also studied assuming the plant to be initially operating at 102 percent of full-power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or the turbine bypass system. The reactor is tripped on the high pressurizer pressure signal. Figures 14.1.10-5 and 14.1.10-6 show the transients with minimum reactivity feedback. The neutron flux remains essentially constant at 102 percent of full power until the reactor is tripped. The DNBR increases throughout the transient. In this case, the pressurizer safety valves are actuated and maintain RCS pressure below 110 percent of the design value. The peak pressurizer pressure is 2554 psia.

Figures 14.1.10-7 and 14.1.10-8 show the transients with maximum reactivity feedback, with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure. The peak pressurizer pressure is 2542 psia.

Conclusions

The analysis indicates that a total loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System and the Steam System. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures. The integrity of the core is maintained by the high pressurizer pressure and low-low steam generator level reactor trips. The minimum DNB ratio for the beginning-of-life case is well above the limit value. At end-of-life the DNB ratio during the total loss of load transient is even higher than that for the steady state, full power operating condition.

Reference

1. Burnett, T W T, et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for Turkey Point Unit 4, Cycle 12, and demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"[1].

Turkey Point Unit 4 operated during Cycle 11 with 61 Westinghouse 15x15 low parasitic (LOPAR) fuel assemblies and 96 Westinghouse 15x15 optimized fuel assemblies (OFA). For Cycle 12 (expected startup March 1989) and subsequent cycles, it is planned to refuel the Turkey Point Unit 4 core with primarily Westinghouse 15x15 optimized fuel assembly (OFA) regions. In a licensing submittal[2] to the NRC, approval was requested for the transition from LOPAR fuel to OFA and associated proposed changes to the Turkey Point Units 3 and 4 Technical Specifications. This licensing submittal justified the compatibility of OFAs and LOPAR fuel assemblies in a mixed-fuel core as well as a full OFA core. The licensing submittal contained mechanical, nuclear, thermal-hydraulic, and accident evaluations which are applicable to the Cycle 12 safety evaluation. Approval of the license application[2] for the OFA transition was granted by the NRC in an SER[3] dated December 9, 1983.

Evaluations have previously been performed by Westinghouse[4] to support the complete or partial removal of thimble plugs from the Turkey Point Units. Safety evaluations for this cycle have been performed such that they remain bounding whether or not thimble plugs are removed from the core.

All of the accidents comprising the licensing bases[6] which could potentially be affected by the fuel reload have been reviewed for the Cycle 12 design described herein. Justification for the applicability of the results of the previous analyses is presented.

1.2 GENERAL DESCRIPTION

The Turkey Point Unit 4 reactor core is comprised of 157 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 11/12 refueling, 52 Region 11 and 12 fuel assemblies will be replaced with 52 fresh Region 14 fuel assemblies. Post Cycle 11 fuel examinations revealed that fuel assembly Z-06 contained one leaking fuel rod at location L-15 and an adjacent fuel rod at location M-15 with deep wear scars, see Figure 4. Assembly Z-06 was reconstituted by replacing the leaking fuel rod, L-15, and the adjacent damaged fuel rod M-15, with filler rods fabricated from stainless steel. The Cycle 12 reload safety evaluation addresses the reconstituted fuel assembly using the standardized methods currently utilized in accepted Westinghouse reload methodology topicals. Existing safety criteria and design limits were applied to the reconstituted fuel assembly including the consideration in the nuclear, thermal-hydraulic and accident analysis of peaking factors and core average linear heat rate effects. In addition, a separate safety evaluation was performed on the reconstituted fuel assembly and is documented in Reference 23. A summary of the Cycle 12 fuel inventory is given in Table 1.

The Cycle 12 core uses 144 Westinghouse Wet Annular Burnable Absorber (WABA) rods. Use of the WABA rods has been generically approved by an NRC SER which is incorporated into the approved version of the Westinghouse WABA evaluation topical[5].

Hafnium burnable absorber rods (240 part-length) will again be used in the core peripheral assemblies in order to reduce the neutron flux at the reactor pressure vessel belt-line weld (see Section 2.1 and Figure 2).

A significant number (2864) of Integral Fuel Burnable Absorber (IFBA) rods will be used for the first time in Turkey Point Unit 4 as part of the Region 14 fuel assemblies. Turkey Point Unit 4 did have demonstration IFBA rods in Cycles 10 and 11. A more detailed description and evaluation of IFBAs is given in References 8 and 9. The NRC has approved the use of IFBAs for Westinghouse 15x15 fuel assemblies in Reference 10.

Nominal core design parameters utilized for the Cycle 12 design are as follows:

Core Power (Mwt)	2200
System Pressure (psia)	2250
Core Inlet Temperature (°F)	546.2
Thermal Design Flow (gpm)	268,500
Average Linear Power Density (kw/ft) (based on 144" active fuel length)	5.58

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 12 design does not cause the previously acceptable safety limits to be exceeded for any accident. This conclusion is based on the following:

1. The actual Cycle 11 burnup is 14,254 MWD/MTU.

2. Cycle 12 burnup is limited to 14,300 MWD/MTU which includes a 500 MWD/MTU power coastdown.

3. There is adherence to plant operating limitations given in the Technical Specifications.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The Region 14 fuel assemblies are Westinghouse 15x15 OFAs which have the same design as the irradiated Region 13D and 13E assemblies from the Cycle 11 core except for the use of: IFBA rods, fuel assemblies modified for extended burnup, reconstitutable top nozzles, standardized fuel pellets, reduced fuel rod backfill pressures, 4g fuel rod plenum springs, and 304L stainless steel top grid sleeve material. The mechanical description and justification of the compatability with the remaining Westinghouse 15x15 LOPAR fuel assemblies in this core are presented in Reference 2. The above design changes are described below and do not affect the safe operation of the Region 14 fuel assemblies.

A total of 2864 IFBA fuel rods were introduced into the Cycle 12 core. These rods have a thin boride coating on the cylindrical surface of the fuel pellets along the central portion (108 inches) of the fuel stack length. In order to offset the effects of the He gas release from the IFBA coating during irradiation, a lower initial He backfill pressure is used in the IFBA rods compared to the non-IFBA fuel rods. Additional information on IFBA coating and rods is given in Reference 8, Addendum 1.

The Region 14 fuel assemblies was modified for extended burnups by reducing the thickness of both the top and bottom nozzle end plates, decreasing the height of the bottom nozzle and increasing the length of the fuel rod. Information on this modified fuel assembly is given in Reference 8.

Details of the Region 14 RTN design features, the design bases and the evaluation of the RTN are given in Section 2.3.2 of Reference 8. Since the RTN design has the same flow area and loss coefficients as the previous design, none of the core/fuel inputs to the safety analysis are affected by the inclusion of the RTN features for Cycle 12 operation. In conjunction with the RTN, a long tapered fuel rod bottom end plug is used to facilitate removal and reinsertion of the fuel rods. The end plug was changed from a chamfered end to a radiused end to improve fuel rod loading and reduce the potential of grid damage during rod loading.

The Region 14 fuel uses the Westinghouse standardized fuel pellet design. This design is a refinement to the previous fuel region's chamfered pellet design with the objective of improving manufacturability while maintaining or improving performance. This design incorporates a reduced pellet length and modifications to the previous chamfer and dish size.

A lower initial helium backfill pressure is used for the non-IFBA Region 14 fuel rods to accomodate extended burnups. Also compared to previous fuel, Region 14 has a smaller fuel rod plenum spring which satisfies a change in the non-operational 6g loading design criteria to "4g axial and 6g lateral loading with dimensional stability." Notification of Westinghouse's plans to generically incorporate this criterion change and the justification of no unreviewed safety question was transmitted to the NRC via Reference 24. The reduced spring force further reduces the already low potential for chamfered pellet chipping in the fuel rod.

The change in grid sleeve material from 304 stainless steel to 304L stainless steel further reduces the already low potential for stress corrosion cracking of the grid sleeves.

Table 1 presents a comparison of pertinent design parameters of the various fuel regions. Region 14A, 14B, 14C, 14D fuel has been designed utilizing the Westinghouse fuel performance model[11] and the Westinghouse clad flattening model.[12] The Westinghouse fuel is designed and operated so that clad flattening will not occur for its planned residence time in the reactor. For all fuel regions, the fuel rod internal pressure design basis, which is discussed and shown acceptable in Reference 13, is satisfied.

Cycle 12 is the second cycle for Unit 4 using the hafnium PTS absorber rods. These rods, which are designed and built by the Exxon Corporation, will be used in core peripheral assemblies (Figure 2) in order to reduce the neutron flux at the reactor pressure vessel belt-line weld. The absorber stack in each of these rods is 36 inches long and is centered 18 inches below the core midplane. Descriptions and dimensions are given in Reference 18. Hafnium absorber rods have also been used in Turkey Point Unit 3 Cycle 11[14].

Rev 7 7/89

For the second time, Unit 4 will use WABA rods which are described and evaluated in the WABA Evaluation Report[5]. The WABA rods have a full-length burnable absorber stack of 134 inches, centered on the core mid-plane.

Westinghouse has had considerable experience with Zircaloy clad fuel. This experience is described in WCAP-8183, "Operational Experience with Westinghouse Cores."[15] Operating experience for Zircaloy grids has also been obtained from six demonstration 17x17 OFAs and four demonstration 14x14 OFAs. This experience is summarized in Attachment B of Reference 2.

2.2 NUCLEAR DESIGN

For the Cycle 12 nuclear design the Westinghouse Advanced Nodal Code (ANC) was introduced to perform the core neutronic analyses and was supplemented with the standard reload methodology design codes given in Reference 1. The ANC code is described in Reference 16 and has received NRC approval, Reference 17. The ANC code incorporated several significant and substantial improvements to the PALADON code used in previous reload designs.

The Cycle 12 loading pattern results in a maximum F_Q less than 2.32 x K(z) at normal operating conditions. Table 2 provides a comparison of the range of values encompassing representative Cycle 12 core kinetics parameters with their current limits based on previously performed accident analyses. The delayed neutron fraction and the prompt neutron lifetime characterize the dynamic response of the core to a change in reactivity. The moderator and Doppler temperature coefficients are measures of the dominant reactivity feedback mechanisms. It can be seen from the Table that all of the Cycle 12 range of values fall within the current limits. Table 3 provides the control rod worths and requirements. The required shutdown margin is based on previously performed accident analyses.[6] The reactivity defects encompass the values for the Cycle 12 core. The available shutdown margin exceeds the minimum required.

The loading contains three different types of discrete burnable absorbers which are described and distributed as follows: 16 Region 14A, and 4 Region 14B, fuel assemblies contain a total of 144 full-length Wet Annular Burnable Absorber (WABA) rods; 8 Region 12A, and 4 Region 12B fuel assemblies contain a total of 240 reduced-length hafnium PTS absorber rods. The hafnium rods are modeled in the Cycle 12 core by approved Westinghouse nuclear design codes, as was done for the Unit 3 Cycle 11 core(14), based on descriptions and dimensions provided to Westinghouse via Reference 18. Burnable absorber lengths and the locations of the secondary source and burnable absorber rods are shown in Figures 2 and 3.

In addition, 2864 IFBAs are distributed in the following fuel regions: 28 Region 14A assemblies (with 60 IFBAs each), 4 Region 14B assemblies (with 116 IFBAs each) and 12 Region 14D assemblies (with 60 IFBAs each). The IFBA rods are reduced length absorbers (108 inches) centered on the fuel midplane.

2.3 THERMAL AND HYDRAULIC DESIGN

The thermal hydraulic methodology, DNBR correlations, and safety analyses used for Cycle 12 are consistent with the OFA transition licensing submittal(2) and the increased $F_{\Delta H}$ limit licensing submittal(21). Based on these methods, no significant variations in thermal margins will result from the Cycle 12 reload. The applicable DNB core safety limits are given in the Technical Specifications, Figure 2.1-1.

The DNB design basis for the hypothetical steamline break event has been changed. The pressures for this event fall in the low pressure range (500-1000 psia) where the W-3 DNB correlation is used for the DNBR design basis with a 1.45 limit DNBR. For previous steambreak analysis, the W-3 correlation was used with a limit DNBR of 1.30. The justification for this new design limit for low pressure applications of the W-3 correlation has been documented by Westinghouse to the NRC.(22) This submittal has not yet been formally accepted by the NRC (although informal agreement has been reached), and formal approval is expected in the near future. However, the hypothetical steamline break event analysis for the Turkey Point Units shows that the above discussed DNBR limit (1.45) criterion is met.

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

This section reviews the plant power capability considering the consequences of those incidents examined in the FSAR using the previously accepted design bases. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% rated power during Cycle 12. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 12 core. The time dependent densification model[19] and the revised Fuel Thermal Safety Model[20] were used for fuel temperature evaluations. The LOCA F_Q limit of 2.32 x K(Z) is met under all operating conditions.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR[6] have been examined. In all cases, it was found that the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis.

The impact of using IFBA rods has been considered. Evaluation of the LOCA accidents has shown that the small increase in IFBA He gas release is accommodated by a reduction in rod internal backfill pressures, and all safety limits are satisfied. For non-LOCA accidents the effects can be accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis.

A safety requirement that the core remain subcritical on soluble boron alone in long term cooling following a Large Break LOCA is adhered to in the Cycle 12 design. This is accomplished by calculating the Post-LOCA core cooling for sources of water that would mix together to provide the resultant core cooling mixture boron concentration. This resultant boron concentration is then compared to the cold zero power critical boron requirement to demonstrate that the core remains subcritical.

A reload can typically affect accident input parameters in three major areas: kinetics characteristics, control rod worths, and core peaking factors. Cycle 12 parameters in each of these areas were examined as discussed below to ascertain whether new accident analyses are required.

3.2.1 Kinetics Parameters

A comparison of Cycle 12 kinetics parameters with current limits is given in Table 2. The delayed neutron fractions, moderator temperature coefficient and prompt neutron lifetime are within the bounds of the current limits.

3.2.2 Control Rod Worths

Changes in control rod worths may affect shutdown margin, differential rod worths, ejected rod worths, and trip reactivity. Table 3 shows that the Cycle 12 shutdown margin requirements are satisfied. As shown in Table 2, the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 12 is less than or equal to the current limit.

3.2.3 Core Peaking Factors

The peaking factors following control rod ejection are within the bounds of the current limits. Evaluation of peaking factors for the rod out of position and dropped RCCA incidents shows that DNBR is maintained above the appropriate safety analysis minimum value.[2,21] For the dropped bank incident, the turbine runback setpoint is sufficient to prevent a DNBR less than the appropriate safety analysis minimum value.[2,21]

4.0 REFERENCES

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3. Letter from McDonald, D. G. (NRC) to Uhrig, R. E. (FP&L), dated December 9, 1983.

4. Ray, S. "Evaluation Report for the Thimble Plug Removal for Turkey Point Units 3 and 4", December 1987.

5. Petrarca, D. J., and Iorii, J. A., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021-P-A (Proprietary) and WCAP-10377-A (Non-Proprietary), October 1983.

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7. Baker, J. S., et al., (Ed.), "Reload Safety Evaluation - Turkey Point Plant Unit 4, Cycle 11 Revision 1," April 1986.

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9. Letter from E. P. Rahe, Jr. (W) to H. Berkow (NRC), NS-NRC-85-3099 "Request for NRC to Review Addendum 1 to WCAP-10445-NP-A", dated December 23, 1985 (Contains Applicability of IFBA design features to W 14x14 and 15x15 fuel assemblies).

10. Letter from C. E. Rossi, (NRC) to E. P. Rahe Jr. (W) "Acceptance for Referencing of Licensing Topical Report, Reference Core Report VANTAGE 5 Fuel Assembly, Addendum 1," dated March 12, 1986.

11. Miller, J. V. (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976.

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13. Risher, D. H. et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963-P-A (Proprietary), August 1978.

14. Skaritka, J., (Ed.), "Reload Safety Evaluation - Turkey Point Plant Unit 3, Cycle 11, Revision 1," April 1987.

15. Skaritka, J., "Operational Experience with Westinghouse Cores," WCAP-8183, Revision 16, August 1988.

16. Liu, Y. S. et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, dated September 1986.

17. "Letter from C. Berlinger (NRC) to E. P. Rahe (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10965-P-A, dated June 23, 1986.

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19. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation," WCAP-8218-P-A, March 1975 (Proprietary) and WCAP-8219-A, March 1975 (Non-Proprietary).

20. Letter from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC), NS-EPR-2673, "Westinghouse Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2, dated October 27, 1982, (Proprietary).

[illegible] [illegible]

[illegible] [illegible] [illegible]

[illegible] [illegible]

21. Letter from Uhrig, R. E. (FP&L) to Eisenhut, D. G., (NRC) "Proposed License Amendment $F_{\Delta H}/F_Q$," Letter No. L-83-455, dated August 19, 1983.

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23. Letter from D. J. Richards (W) to S. T. Hale (FP&L), "Florida Power and Light Co., Turkey Point Unit 4, Cycle 12, Fuel Assembly Reconstitution Safety Evaluation," FPL-88-958, dated December 6, 1988.

24. Letter from E. P. Rahe, Jr. (W) to L. E. Phillips (NRC) "Fuel Handling Load Criteria (6g vs. 4g)" NS-EPR-2893, dated April 12, 1984.

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS

TURKEY POINT UNIT 4 - CYCLE 12

Region	9B	11B	12A	12B	12C	13A(a)	13B(b)
Enrichment (w/o U 235)*	3.396	3.398	2.606	3.447	3.010	3.014	3.101
Density (% Theoretical)*	94.48	94.46	94.95	95.03	94.90	95.06	94.56
Number of Assemblies	1	8	8	12	8	4	4
Approximate Burnup at Beginning of Cycle 12 (MWD/MTU)+	16100	29200	22700	28300	24800	18500	17600
Fuel Type	LOPAR	LOPAR	OFA	OFA	LOPAR	LOPAR	LOPAR
Number of IFBA Fuel Rods	--	--	--	--	--	--	--

*All fuel region values are as-built
+Based on Cycle 11 burnup of 14254 MWD/MTU, rounded to the nearest 100 MWD/MTU

(a) Was originally designated Region 12C (3 assemblies) and Region 11C (1 assembly) for Unit 4
(b) Was originally designated Region 11A for Unit 4

TABLE 1 (Continued)

FUEL ASSEMBLY DESIGN PARAMETERS

TURKEY POINT UNIT 4 - CYCLE 12

Region	13C(c)	13D	13E	14A	14B	14C	14D
Enrichment (w/o U 235)*	3.124	3.196	3.404	3.402	3.402	3.797	3.797
Density (% Theoretical)*	94.53	94.85	95.36	95.50	95.50	95.23	96.23
Number of Assemblies	4	24	32	28	4	8	12
Approximate Burnup at Beginning of Cycle 12 (MWD/MTU)+	17600	18800	15200	0	0	0	0
Fuel Type	LOPAR	OFA	OFA	OFA	OFA	OFA	OFA
Number of IFBA Fuel Rods	--	--	--	1680	464	--	720

*All fuel region values are as-built
+Based on Cycle 11 burnup of 14254 MWD/MTU

(c) Was originally designated Region 12D for Unit 4

TABLE 2

KINETICS CHARACTERISTICS
TURKEY POINT UNIT 4 - CYCLE 12

	Current Limit(2)(7)	Cycle 12
Moderator Temperature Coefficient (pcm/°F)*		
a. most positive	+5.0 (≤ 70% RTP), linear ramp to 0 at 100% RTP	+5.0 (≤ 70% RTP), linear ramp to 0 at 100% RTP**
b. most negative ***	-50	>-50
Doppler Coefficient (pcm/°F)	-2.9 to -1.0	-2.9 to -1.0
Delayed Neutron Fraction βeff, (%)	0.44 to 0.75	0.44 to 0.75
Maximum Prompt Neutron Lifetime, (μ sec)	26	<26
Maximum Differential Rod Worth of Two Banks Moving Together at HZP (pcm/in)*	100	<100

* pcm = $10^{-5}\ \Delta\rho$

** Although it is highly unlikely for this design, the moderator temperature coefficient for the all-rods-out condition may be more positive than the current limit at BOC. The moderator temperature coefficient will be kept within the MTC limit by administrative controls (with the appropriate D bank position and/or boron concentration).

*** All rods in condition, corresponds to a most positive moderator density coefficient (MDC) of 0.43 Δρ/gm/cc. The most positive MDC is incrementally corrected to nominal operating conditions to obtain the most negative MTC Technical Specification limit of -35 pcm/°F. This correction involves subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and a conversion for the rate of change of moderator density with temperature at Rated Thermal Power conditions.

TABLE 3

SHUTDOWN REQUIREMENTS AND MARGINS
TURKEY POINT UNIT 4 - CYCLES 11 AND 12

	Cycle 11		Cycle 12	
	BOC	EOC	BOC	EOC
Control Rod Worth (% $\Delta\rho$)				
All Rods Inserted Less Worst Stuck Rod	6.51	6.43	6.41	6.53
(1) Less 10%	5.86	5.77	5.77	5.88
Control Rod Requirements (%$\Delta\rho$)				
Reactivity Defects (Doppler, T_{avg}, Void, Redistribution)	1.74	3.12	1.90	3.27
Rod Insertion Allowance	1.53	0.50	1.52	0.50
(2) Total Requirements	3.27	3.62	3.42	3.77
Shutdown Margin (1)-(2) (%$\Delta\rho$)	2.59	2.17	2.35	2.11
Required Shutdown Margin (%$\Delta\rho$)	1.00	1.77	1.00	1.77

FIGURE 1
TURKEY POINT UNIT 4 CYCLE 12
REFERENCE CORE LOADING PATTERN

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R							12A	12B	12A						
P					11B	13E	14C	13B	14C	13E	11B				
N				12C	14D	14A	13D	14A	13D	14A	14D	12C			
M			12C	14D	13E	13D	14A	12B	14A	13D	13E	14D	12C		
L		11B	14D	13E	13C	14A	13D	13E	13D	14A	13C	13E	14D	11B	
K		13E	14A	13D	14A	12B	13E	13E	13E	12B	14A	13D	14A	13E	
J	12A	14C	13D	14A	13D	13E	14B	13A	14B	13E	13D	14A	13D	14C	12A
H	12B	13B	14A	12B	13E	13E	13A	9B	13A	13E	13E	12B	14A	13B **	12B
G	12A	14C	13D	14A	13D	13E	14B	13A	14B	13E	13D	14A	13D	14C	12A
F		13E	14A	13D	14A	12B	13E	13E	13E	12B	14A	13D	14A	13E	
E		11B	14D	13E	13C	14A	13D	13E	13D	14A	13C	13E	14D	11B	
D			12C	14D	13E	13D	14A	12B	14A	13D	13E	14D	12C		
C				12C	14D	14A	13D	14A	13D	14A	14D	12C			
B					11B	13E	14C	13B	14C	13E	11B				
A							12A	12B	12A						

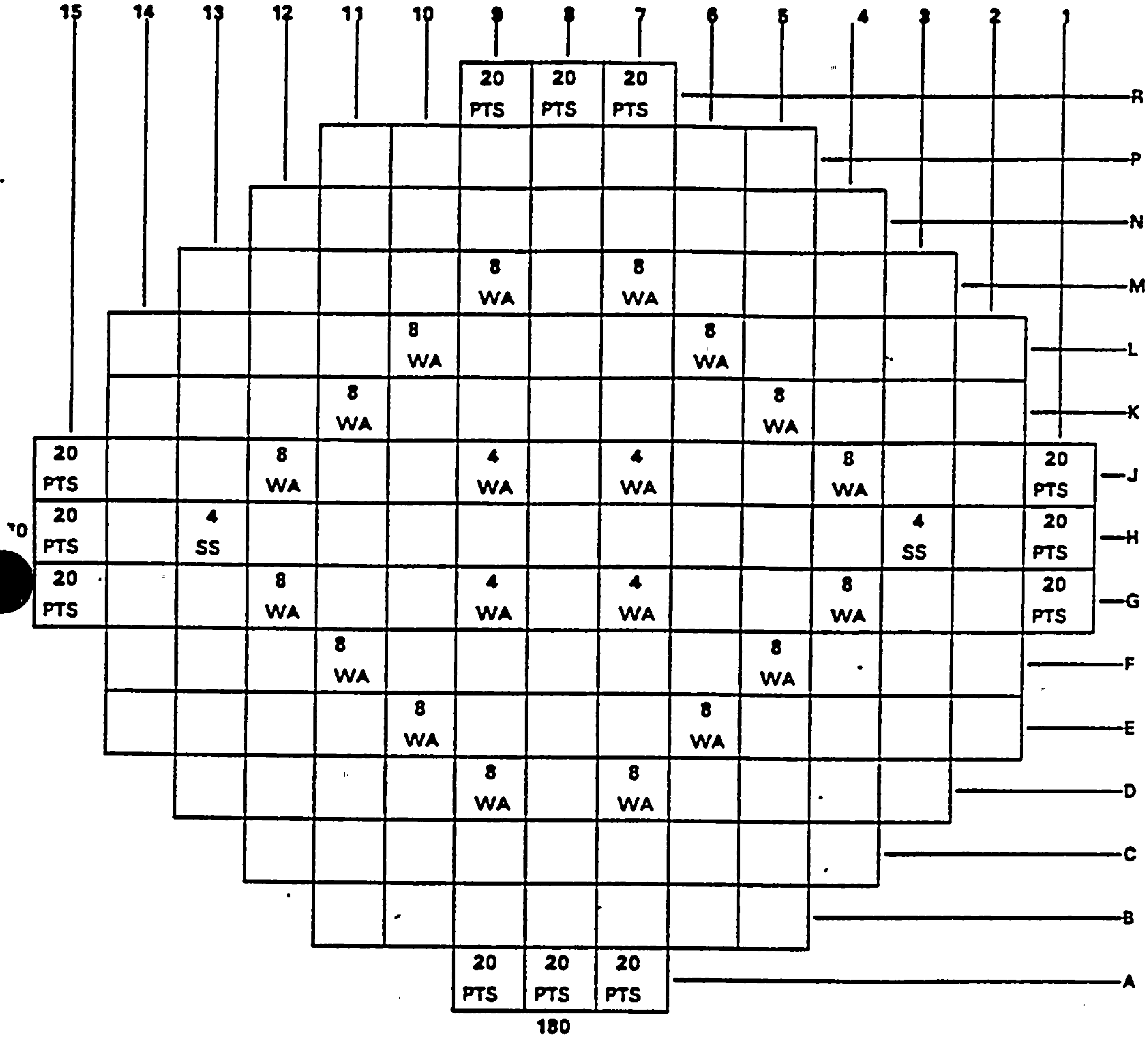
70

180

**. Assembly Z-06: Contains 2 Stainless Steel Rods (see Figure 4)

Rev 7 7/89

FIGURE 2
TURKEY POINT UNIT 4 CYCLE 12
DISCRETE ABSORBER, SECONDARY SOURCE LOCATIONS



N XX Number of Components
Type of Components

SS = Secondary Sources
WA = Full Length Wet Annular Burnable Absorbers
PTS = Reduced Length Hafnium Absorber Rods

FIGURE 3
TURKEY POINT UNIT 4 CYCLE 12
INTEGRAL FUEL BURNABLE ABSORBER LOCATIONS

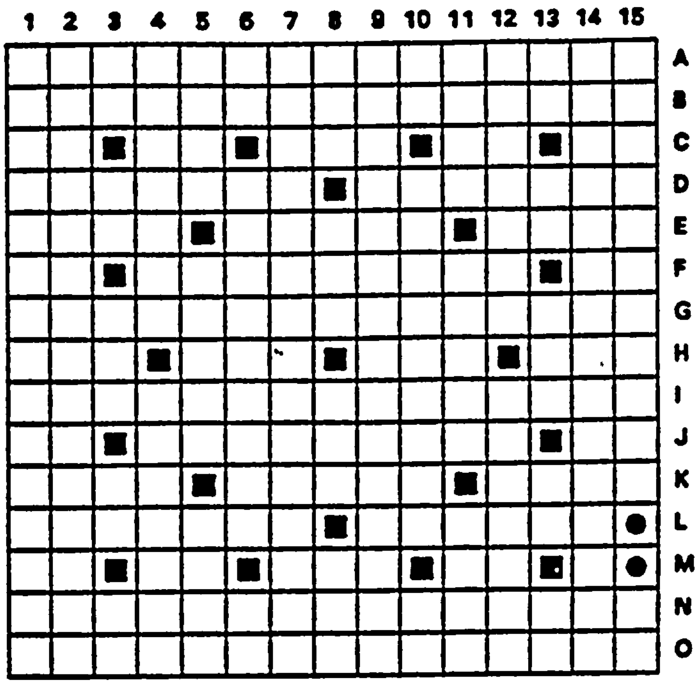
	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R															
P															
N					60 IFBA	60 IFBA		60 IFBA		60 IFBA	60 IFBA				
M				60 IFBA			60 IFBA		60 IFBA			60 IFBA			
L			60 IFBA			60 IFBA				60 IFBA			60 IFBA		
K			60 IFBA		60 IFBA						60 IFBA		60 IFBA		
J				60 IFBA			116 IFBA		116 IFBA			60 IFBA			
H			60 IFBA										60 IFBA		
G				60 IFBA			116 IFBA		116 IFBA			60 IFBA			
F			60 IFBA		60 IFBA						60 IFBA		60 IFBA		
E			60 IFBA			60 IFBA				60 IFBA			60 IFBA		
D				60 IFBA			60 IFBA		60 IFBA			60 IFBA			
C					60 IFBA	60 IFBA		60 IFBA		60 IFBA	60 IFBA				
B															
A															

270 (left side, row H)

180 (bottom)

XX / IFBA: Number of IFBA, Integral Fuel Burnable Absorber

FIGURE 4
TURKEY POINT UNIT 4 CYCLE 12
FUEL RECONSTITUTION FOR ASSEMBLY Z-06



Assembly Z-06

Fuel Rod

Stainless Steel Rod

Guide Tube or Instrumentation Thimble

TABLE 4.1-3

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design/Operating Temperature, ^{o}F.	680/653
Water Volume, Full Power, ft. ^{3}*	780
Steam Volume, Full Power, ft. ^{3}	520
Surge Line Nozzle Diameter, in./Pipe Schedule	14/Sch 140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw (total)	1300
Heatup rate of Pressurizer using Heaters only, ^{o}F/hr.	55 (approximately)
Power Relief Valves: #455C & 456	
Number	2
Set Pressure (open), psig	
i) Normal operation	2335
ii) OMS Actuation during Heatup or Cooldown	
a) RCS $\lesssim$ 285^{o}F	415 ±15
b) RCS > 285^{o}F.	Setpoint increases step-wise to 2335 psig as temperature increases to 750^{o}F.
Capacity, lb/hr saturated steam/valve	210,000
Safety Valves	
Number	3
Set Pressure, psig	2485±1%
Capacity, lb/hr saturated steam/valve	293,330

Pressurizer Relief Tank

Design pressure, psig	100
Rupture disc release pressure, psig	100
Design temperature, ^{o}F.	340
Normal water temperature, ^{o}F.	120
Total volume, ft^{3}	1300
Rupture disc relief capacity, lb/hr	900,000

* 60% of net internal volume (maximum calculated power)

Rev. 3-7/85

7.2 PROTECTIVE SYSTEMS

The protection systems consists of the control and instrumentation associated with the Engineered Safety Features and the Reactor Protection System.

7.2.1 DESIGN BASES

Core Protection Systems

Criterion: Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. (GDC 14)

If the reactor protection system receives signals which are indicative of an approach to unsafe operating conditions, the system actuates alarms, prevents control rod withdrawal, initiates load cutback, and/or opens the reactor trip breakers.

The basic reactor operating philosophy is to define an allowable region of power and coolant temperature conditions. This allowable range is defined by the primary tripping functions, the overpower ΔT trip, the over-temperature ΔT trip and the nuclear overpower trip. The operating region below these trip settings is designed so that no combination of power, temperatures and pressure could result in DNBR less than 1.30 with all reactor coolant pumps in operation. A complete list of tripping functions may be found in Table 7.2-1.

This section contains Figures 7.2-5, 7.2-8a, 7.2-8b, and 7.2.8c which list nominal setpoint values for illustrative purposes. The actual limiting setpoint values are contained in the Technical Specifications.