April 23, 1991

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Docket Nos. 50-250
and 50-251

Posted

Amdt. 135 to DPR-41

Mr. J. H. Goldberg
President-Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Goldberg:

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS RE: REMOVAL OF RTD BYPASS MANIFOLD (TAC NOS. 77633 AND 77634)

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-31 and Amendment No. 135 to Facility Operating License No. DPR-41 for the Turkey Point Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated September 13, 1990. Supplemental information was provided in letters dated November 15, 1990, December 13, 1990, January 28, 1991.

These amendments provide for replacement of the resistance thermal detector (RTD) bypass manifold system with fast-response thermowell-mounted RTDs. The amendments also provide for use of EAGLE-21 digital electronics in the reactor protection and control systems, and change the surveillance interval and allowed testing "Bypass" time for the new electronic system. Finally, the amendments include modifications to the axial power imbalance term in the overtemperature/delta-temperature and overpower/delta-temperature reactor trip functions.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
(Original Signed By)

Rajender Auluck, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.140 to DPR-31
2. Amendment No.135 to DPR-41
3. Safety Evaluation

cc w/enclosures:
See next page

C	:LA:PD22	:PM:PD22	:D:PD22	:OGC	: PM:PD22	:	:
NAME	:DMiller	:RAuluck:kdj	:HBerkow	:RBachmann	: JWilliams	:	:
DATE	:4/3/91	:4/5/91	:4/12/91	:4/8/91	:4/23/91	:	:

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Document Name: AMEND TP 77633/34

Mr. J. H. Goldberg
Florida Power and Light Company

Turkey Point Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. DPR-31

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated September 13, 1990, as supplemented by letters dated November 15, 1990, December 13, 1990, January 4, 1991, and January 28, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

   (B) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 140, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance. The upgrades and enhancements associated with this amendment will be implemented prior to entry into Mode 3. It is understood that resistance thermal detector cross-calibration and response time tests, and a reactor coolant system leak test will be performed following entry into Mode 3. Additionally, a flow calorimetric measurement will be performed upon achieving stable full power operation. All other testing required to demonstrate proper operation of modified components will be completed prior to entry into Mode 3.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 23, 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 135
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated September 13, 1990, as supplemented by letters dated November 15, 1990, December 13, 1990, January 4, 1991, and January 28, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

   (B) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 135, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance. The upgrades and enhancements associated with this amendment will be implemented prior to entry into Mode 3. It is understood that resistance thermal detector cross-calibration and response time tests, and a reactor coolant system leak test will be performed following entry into Mode 3. Additionally, a flow calorimetric measurement will be performed upon achieving stable full power operation. All other testing required to demonstrate proper operation of modified components will be completed prior to entry into Mode 3.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 23, 1991

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 140 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 135 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-6	1-6
2-3 to 2-10	2-3 to 2-10
B2-3	B2-3
- -	B2-3a
B2-4	B2-4
B2-5	B2-5
3/4 3-2	3/4 3-2
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-15	3/4 3-15
3/4 3-18	3/4 3-18
3/4 3-22 to 3/4 3-34*	3/4 3-22 to 3/4 3-34*
B 3/4 3-1	B 3/4 3-1
- -	B 3/4 3-1a

*Pages 3/4 3-32, 3/4 3-32a and 3/4 3-33a did not have any changes in content. Due to the reformatting of the table, the pages had to be renumbered.

DEFINITIONS

THERMAL POWER

1.31 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.32 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

DIGITAL CHANNEL OPERATIONAL TEST

1.37 A DIGITAL CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock, and/or trip functions.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the setpoint consistent with the Trip setpoint value within permissible calibration tolerance.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

EQUATION 2.2-1 $Z + R + S \leq TA$

where:

Z = The value for column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value of Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance in % of span) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE #
1. Manual Reactor Trip	N.A	N.A	N.A	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	[ ]	[ ]	[ ]	≤109% of RTP**	≤[ ]% of RTP**
b. Low Setpoint	[ ]	[ ]	[ ]	≤25% of RTP**	≤[ ]% of RTP**
3. Intermediate Range, Neutron Flux	[ ]	[ ]	[ ]	≤25% of RTP**	≤[ ]% of RTP**
4. Source Range, Neutron Flux	[ ]	[ ]	[ ]	≤10^5 cps	≤[ ] x 10^5 cps
5. Overtemperature ΔT	7.2	4.8	3.0	See Note 1	See Note 2
6. Overpower ΔT	5.3	3.1	2.0	See Note 3	See Note 4
7. Pressurizer Pressure-Low	[ ]	[ ]	[ ]	≥1835 psig	≥[ ] psig
8. Pressurizer Pressure-High	[ ]	[ ]	[ ]	≤2385 psig	≤[ ] psig
9. Pressurizer Water Level-High	8.0	6.8	4.0	≤92% of instrument span	≤92.2% of instrument span
10. Reactor Coolant Flow-Low	4.6	2.7	0.8	≥90% of loop design flow*	≥88.7% of loop design flow*
11. Steam Generator Water Level Low-Low	[ ]	[ ]	[ ]	≥15% of narrow range instrument span	≥[ ]% of narrow range instrument span

*Loop design flow = 89,500 gpm
**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE #
12. Steam/Feedwater Flow Mismatch Coincident With	[ ]	[ ]	[ ]	Feed Flow $\leq 0.64 \times 10^6$ lb/hr below steam flow	Feed Flow $\leq$[ ] $\times 10^6$ lb/hr below steam flow
Steam Generator Water Level-Low	[ ]	[ ]	[ ]	$\geq$15% of narrow range instrument span	$\geq$[ ]% of narrow range instrument span
13. Undervoltage - 4.16 kV Busses A and B	[ ]	[ ]	[ ]	$\geq$2496 volts-each bus	$\geq$[ ] volts-each bus
14. Underfrequency - Trip of Reactor Coolant Pump Breaker(s) Open	[ ]	[ ]	[ ]	$\geq$56.1 Hz	$\geq$[ ] Hz
15. Turbine Trip					
a. Auto Stop Oil Pressure	[ ]	[ ]	[ ]	$\geq$45 psig	$\geq$[ ] psig
b. Turbine Stop Valve Closure	N.A	N.A	N.A	Fully Closed ***	Fully Closed ***
16. Safety Injection Input from ESF	N.A	N.A	N.A	N. A.	N.A.
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	[ ]	[ ]	[ ]	$\geq 1 \times 10^{-10}$ amp	$\geq$[ ] amp

***Limit switch is set when Turbine Stop Valves are fully closed.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE #
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	[ ]	[ ]	[ ]	$\leq$10% of RTP**	$\leq$[ ]% of RTP**
2) Turbine First Stage Pressure	[ ]	[ ]	[ ]	$\leq$10% Turbine Power	$\leq$[ ]% Turbine Power
c. Power Range Neutron Flux, P-8	[ ]	[ ]	[ ]	$\leq$45% of RTP**	$\leq$[ ]% of RTP**
d. Power Range Neutron Flux, P-10	[ ]	[ ]	[ ]	$\geq$10% of RTP**	$\geq$[ ]% of RTP**
18. Reactor Coolant Pump Breaker Position Trip	N.A	N.A	N.A	N.A.	N.A.
19. Reactor Trip Breakers	N.A	N.A	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A	N.A	N.A	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left\{\frac{1+\tau_1 S}{1+\tau_2 S}\right\} \left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)}\left[T\left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Instrumentation
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead/Lag compensator on measured ΔT; τ_1=8s, τ_2 =3s
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT; τ_3 = 0s
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
K_1	=	1.095;
K_2	=	0.0107/°F;
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 25s, τ_5 = 3 s;
T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg}; τ_6 = 0s
T'	$\le$	574.2°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000453/psig;
P	=	Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and f_2 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between - 14% and + 10%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 14%, the ΔT Trip Setpoint shall be automatically reduced by 1.5% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 10%, the ΔT Trip Setpoint shall be automatically reduced by 1.5% of its value at RATED THERMAL POWER.

NOTE 2: The channels maximum trip setpoint shall not exceed its computed setpoint by more than 1.5% of instrument span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \left\{\frac{1 + \tau_1 S}{1 + \tau_2 S}\right\} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 \left[T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''\right] - f_2 \ (\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	1.09,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00068/°F for T > T" and K_6 = 0 for T ≤ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 574.2°F),

S = As defined in Note 1, and

f_2 (ΔI) = 0 for all ΔI

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 1.4% of instrument span.

If no allowable value and no allowance (TA), Z, or S is specified as indicated by [ ], the trip setpoint shall also be the allowable value.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the specified Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. If no value is listed in the Allowable column, the setpoint value is the limiting setting.

For some functions, an optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints includes an allowance for instrument uncertainties. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations for all power levels to mitigate the consequences of a reactivity excursion which may be too rapid for the temperature and pressure protective trips.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit is taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors and pressure is within the range between the Pressurizer High and Low Pressure trips. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower ΔT

The Overpower ΔT trip prevents power density anywhere in the core from exceeding 118% of the design power density. This provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine first stage pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer Water Level-High trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine first stage pressure at approximately 10%

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ACTUATION LOGIC TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 13 - With the number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1***, 2
3. Intermediate Range, Neutron Flux	S	R(4)	S/U(1),M	N.A.	N.A.	1***, 2
4. Source Range, Neutron Flux	S	R(4)	S/U(1),M(9)	N.A.	N.A.	2**, 3, 4, 5
5. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
6. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
7. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	1
8. Pressurizer Pressure--High	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
10. Reactor Coolant Flow--Low	S	R	M	N.A.	N.A.	1
11. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(12) NOT USED.

(13) Remote manual undervoltage trip when breaker placed in service.

(14) Interlock Logic Test shall consist of verifying that the interlock is in its required state by observing the permissive annunciator window.

(15) Automatic undervoltage trip.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value within permissible calibration tolerance.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

EQUATION 2.2-1 $$Z + R + S \leq TA$$

where:

Z = The value for column Z of Table 3.3-3 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value of Column S (Sensor Error) of Table 3.3-3 for the affected channel, and

TA = The value from Column TA (Total Allowance in % of span) of Table 3.3-3 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam line in any two steam lines	1/steam generator in any two steam lines	1, 2, 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with:	2/steam line	1/steam line	1/steam line	1, 2, 3	15
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater###					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, comply with Specification 3.0.3.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the control room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 25 - With the number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Ventilation Isolation, Start Diesel Generators, Containment Phase A Isolation (except Manual SI), Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water)					
a. Manual Initiation	N.A	N.A	N.A	N.A.	N.A.
b. Automatic Actuation Logic	N.A	N.A	N.A	N.A.	N.A.
c. Containment Pressure--High	[ ]	[ ]	[ ]	≤6 psig	≤[ ] psig
d. Pressurizer Pressure--Low	[ ]	[ ]	[ ]	≥1715 psig	≥[ ] psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	[ ]	[ ]	[ ]	≤150 psi	≤[ ] psi
f. Steam Line Flow--High	[ ]	[ ]	[ ]	≤A function defined as follows: A Δp corresponding to 0.64×10^6 lbs/hr at 0% load increasing linearly to a Δp corresponding to 3.84×10^6 lbs/hr at full load.	[ ]

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
Coincident with: Steam Generator Pressure--Low	[ ]	[ ]	[ ]	≥600 psig	≥[ ] psig
or					
T_{avg}--Low	4.0	2.0	1.0	≥543°F	≥542.5°F
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	N.A	N.A	N.A	N.A.	N.A.
b. Containment Pressure--High-High Coincident with:	[ ]	[ ]	[ ]	≤30.0 psig	≤[ ] psig
Containment Pressure--High	[ ]	[ ]	[ ]	≤6.0 psig	≤[ ] psig
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	see item 1			See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.	
b. Phase "B" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-High Coincident with:	[ ]	[ ]	[ ]	≤30.0 psig	≤[ ] psig
Containment Pressure--High	[ ]	[ ]	[ ]	≤6.0 psig	≤[ ] psig
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
4) Containment Radio-activity--High (1)	[ ]	[ ]	[ ]	Particulate (R-11) ≤6.1×10^5 CPM Gaseous (R-12) See (2)	[ ]
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
4. Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-High Coincident with:	[ ]	[ ]	[ ]	≤30.0 psig	≤[ ] psig
Containment Pressure--High	[ ]	[ ]	[ ]	≤6.0 psig	≤[ ] psig
f. Steam Line Flow--High	[ ]	[ ]	[ ]	≤A function defined as follows: A Δp corresponding to 0.64×10^6 lbs/hr at 0% load increasing linearly to a Δp corresponding to 3.84×10^6 lbs/hr at full load.	[ ]
Coincident with: Steam Line Pressure--Low	[ ]	[ ]	[ ]	≥600 psig	≥[ ] psig
or T_{avg}--Low	4.0	2.0	1.0	≥543°F	≥542.5°F
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A.	N.A.	N.A.
b. Safety Injection	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
6. Auxiliary Feedwater (3)					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	[ ]	[ ]	[ ]	≥15% of narrow range instrument span.	≥[ ]% of narrow range instrument span.
c. Safety Injection	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
d. Bus Stripping	see item 7			See Item 7. below for all Bus Stripping Setpoints and Allowable Values.	
e. Trip of All Main Feedwater Pump Breakers.	N.A.	N.A	N.A	N.A.	N.A.
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	N.A	N.A	N.A	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
7. Loss of Power (Continued)					
b. 480V Load Centers (Instantaneous Relays) Degraded Voltage					
Load Center					
3A	[ ]	[ ]	[ ]	418V±5V (10 sec ± 1 sec delay)	[ ]
3B	[ ]	[ ]	[ ]	423V±5V (10 sec ± 1 sec delay)	[ ]
3C	[ ]	[ ]	[ ]	429V±5V (10 sec ± 1 sec delay)	[ ]
3D	[ ]	[ ]	[ ]	429V±5V (10 sec ± 1 sec delay)	[ ]
4A	[ ]	[ ]	[ ]	407V±5V (10 sec ± 1 sec delay)	[ ]
4B	[ ]	[ ]	[ ]	423V±5V (10 sec ± 1 sec delay)	[ ]
4C	[ ]	[ ]	[ ]	419V±5V (10 sec ± 1 sec delay)	[ ]
4D	[ ]	[ ]	[ ]	404V±5V (10 sec ± 1 sec delay)	[ ]
Coincident with: Safety Injection and	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
Diesel Generator Breaker Open				N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
7. Loss of Power (Continued)					
c. 480V Load Centers (Inverse Time Relays) Degraded Voltage					
Load Center					
3A	[ ]	[ ]	[ ]	416V±5V(60 sec ±30 sec delay)	[ ]
3B	[ ]	[ ]	[ ]	426V±5V(60 sec ±30 sec delay)	[ ]
3C	[ ]	[ ]	[ ]	436V±5V(60 sec ±30 sec delay)	[ ]
3D	[ ]	[ ]	[ ]	437V±5V(60 sec ±30 sec delay)	[ ]
4A	[ ]	[ ]	[ ]	424V±5V(60 sec ±30 sec delay)	[ ]
4B	[ ]	[ ]	[ ]	422V±5V(60 sec ±30 sec delay)	[ ]
4C	[ ]	[ ]	[ ]	433V±5V(60 sec ±30 sec delay)	[ ]
4D	[ ]	[ ]	[ ]	432V±5V(60 sec ±30 sec delay)	[ ]
Coincident with: Diesel Generator Breaker Open	N.A.	N.A.	N.A	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
8.	Engineering Safety Features Actuation System Interlocks					
a.	Pressurizer Pressure	[ ]	[ ]	[ ]	≤2000 psig	≤[ ] psig
b.	T_{avg}--Low	4.0	2.0	1.0	≥543°F	≥542.5 °F
9.	Control Room Ventilation Isolation					
a.	Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Safety Injection	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
c.	Containment Radioactivity--High (1)	[ ]	[ ]	[ ]	Particulate (R-11) <6.1 x 10^5 CPM Gaseous (R-12) See (2)	[ ]
d.	Containment Isolation Manual Phase A or Manual Phase B	N.A	N.A	N.A	N.A.	N.A.
e.	Air Intake Radiation Level	[ ]	[ ]	[ ]	≤ 2 mR/hr	2.83 mR/hr

TABLE NOTATIONS

(1) Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (continued)

(2) Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(F)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in Specification 3.11.2.1.

(3) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

#If no allowable value, ALLOWANCE (TA), Z or S is specified so indicated by [ ], the trip setpoint shall also be the allowable value.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Ventilation Isolation, Start Diesel Generators, Containment Phase A Isolation (except Manual SI), Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water)						
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3(3)
c. Containment Pressure--High	N.A.	R	N.A.	N.A.	M(1)	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M(5)	N.A.	N.A.	1, 2, 3(3)
e. High Differential Pressure Between the Steam Line Header and any Steam Line	S	R	M(5)	N.A.	N.A.	1, 2, 3(3)
f. Steam Line Flow--High Coincident with:	S	R	M(5)	N.A.	N.A.	1, 2, 3(3)
Steam Generator Pressure--Low	S	R	M(5)	N.A.	N.A.	1, 2, 3(3)
or						
T_{avg}--Low	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Containment Spray						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Containment Pressure--High-High	N.A.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
3. Containment Isolation						
a. Phase "A" Isolation						
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
b. Phase "B" Isolation						
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)						
3) Containment Pressure--High-High Coincident	N.A.	R	N.A.	R	M(1)	1, 2, 3
with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
c. Containment Ventilation Isolation						
1) Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
4) Containment Radioactivity--High	S	R	M	N.A.	N.A.	1, 2, 3, 4
4. Steam Line Isolation						
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3(3)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation (Continued)						
c. Containment Pressure--High-High	N.A.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
d. Steam Line Flow--High	S(3)	R	M(5)	N.A.	N.A.	1, 2, 3
Coincident with: Steam Generator Pressure--Low	S(3)	R	M(5)	N.A.	N.A.	1, 2, 3
or T_{avg}--Low	S(3)	R	Q(5)	N.A.	N.A.	1, 2, 3
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	1, 2
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
6. Auxiliary Feedwater (2)						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)						
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
d. Bus Stripping	N.A.	R	N.A.	R	N.A.	1, 2, 3
e. Trip of All Main Feedwater Pump Breakers.	N.A.	N.A.	N.A.	R	N.A.	1, 2
7. Loss of Power						
a. 4.16 kV Busses A and B (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	1, 2, 3, 4
b. 480V Load Centers 3A,3B,3C,3D and 4A,4B,4C,4D (Instantaneous Relays) Degraded Voltage	S	R	N.A.	M(1)	N.A.	1, 2, 3, 4
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. 480V Load Centers 3A,3B,3C,3D and 4A,4B,4C,4D (Inverse Time Relays) Degraded Voltage	S	R	N.A.	M(1)	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Engineering Safety Features Actuation System Interlocks						
a. Pressurizer Pressure	N.A.	R	M(5)	N.A.	N.A.	1, 2, 3(3)
b. T_{avg}--Low	N.A.	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
9. Control Room Ventilation Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Containment Radioactivity--High	S	R	M	N.A.	N.A.	(4)
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
e. Control Room Air Intake Radiation Level	S	R	M	N.A.	N.A.	All

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.
(3) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.
(4) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.
(5) Test of alarm function not required when alarm locked in.

#At least once per 18 months each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance (due to plant specific design, pulling fuses and using jumpers may be used to place channels in trip), and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Under some pressure and temperature conditions, certain surveillances for Safety Injection cannot be performed because of the system design. Allowance to change modes is provided under these conditions as long as the surveillances are completed within specified time requirements.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-3 are the nominal values at which the bistables are set for each functional unit. The setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-3. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. If no value is listed in the Allowable column, the Setpoint value is the limiting setting.

For some functions, an optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured"

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

values of the errors are considered. Z, as specified in Table 3.3-3, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-3, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints includes an allowance for instrument uncertainties. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO.135 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By letter dated September 13, 1990, Florida Power and Light Company (FPL or the licensee), requested amendments to Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4, respectively. Supplemental information was provided by letters dated November 15, 1990, December 13, 1990, January 4, 1991, and January 28, 1991. The proposed amendments would change the Technical Specifications (TS) to accommodate replacement of the resistance thermal detector (RTD) bypass manifold system with fast-response thermowell-mounted RTDs. Replacing the RTD bypass system is expected to improve system reliability and reduce personnel radiation exposure.

The proposed changes would include elimination of the f(delta-I) axial power imbalance term from the overpower/delta-temperature (OPdT), and reduction in the f(delta-I) slope for the overtemperature/delta-temperature (OTdT) reactor trip functions. The proposed amendments would also include use of upgraded electronics in the reactor protection and control systems, and would change the surveillance interval and allowed testing "Bypass" time for the new electronic system. The January 4 and 28, 1991 letters provided clarifying information which did not change the initial proposed no significant hazards consideration determination.

The licensee proposes to implement these changes during the current dual-unit outage for emergency power system upgrades.

2.0 EVALUATION

Staff review of the proposed design modifications is based upon guidance from Sections 7.2 and 7.3 of the Standard Review Plan (NUREG-0800). Review objectives are to confirm that the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS) continue to satisfy the applicable acceptance criteria and guidelines, and that the systems will perform their safety functions for all required conditions.

2.1 RTD Bypass System Removal

2.1.1 Current System

The current method of measuring the hot and cold leg reactor coolant temperatures uses an RTD bypass system. The hot and cold leg temperature readings from each coolant loop are used for protection and control system inputs. The RTD bypass system was designed to address temperature streaming (non-uniform stratified flow in the cross-section) in the hot legs and, by use of shutoff valves, to allow replacement of the direct immersion narrow-range RTDs without draindown of the Reactor Coolant System (RCS). For increased accuracy in measuring the hot leg temperatures, sampling scoops were placed in each hot leg at three locations of a cross-section, 120 degrees apart. Each scoop has five orifices which sample the hot leg flow along the leading edge of the scoop. The flow from the scoops is piped to a manifold where a direct immersion RTD measures the average hot leg temperature of the flow from the three scoops in each hot leg. This bypass flow is routed back downstream of the steam generator. The cold leg temperature is measured in a similar manner except that no scoops are used, as temperature streaming is not a problem due to the mixing action of the RCS pump.

2.1.2 Proposed System

The licensee will replace the direct-immersion RTDs with Weed Instrument Co., Inc. dual-element RTDs mounted in thermowells. The spare element in each RTD assembly will be terminated at the containment penetration.

The licensee proposes to remove all the bypass piping and associated valves and hangers. The bypass connections that return RCS coolant back to the cold leg crossover piping will be capped. Where possible, the scoops in the hot legs will be modified to accept RTD thermowells, which will allow RTD replacement without requiring draindown of the RCS. If structural components interfere with the placement of a thermowell in an existing scoop or nozzle, the licensee will cap the scoop or nozzle and prepare an alternate penetration to accommodate an RTD thermowell. The Turkey Point units are 3-loop Westinghouse plants, and this work will affect all three loops of each unit.

The hot leg RTDs (three in each hot leg) and the cold leg RTDs (one in each cold leg) will be connected to a newly installed Westinghouse Eagle 21 Temperature Averaging System (TAS). In this system, for each loop, each hot leg temperature signal, T(hot), is first subjected to a range check. Then an estimated average hot leg temperature, T(est,j), is derived from each T(hot) by applying a temperature streaming correction bias, S(j). The TAS then uses the resulting T(est,j) signals to calculate an estimated average hot leg temperature for the corresponding loop, T(est,ave). The three T(est,j)s are then compared to this T(est,ave) to determine whether they agree within a specified temperature range (±DELTAH). If the T(est,j) signals agree within ±DELTAH of T(est,ave), the group quality is set to GOOD, and the loop average hot leg temperature, T(hot,ave), is set to the average of the three estimated average hot leg temperatures.

If a T(est,j) signal does not agree within ±DELTAH of T(est,ave), the signal value with the greatest deviation from T(est,ave) is deleted and the quality of the deleted signal is set to POOR. The remaining signals are then checked for consistency (±DELTAH). If the two signals pass the consistency check (within ±DELTAH), the group value, T(hot, ave), is set to the average of the two signals, and the group quality is set to POOR. If the two remaining signals are not consistent, the T(hot,ave) value is set to the average of the two signals, and the group quality is set to BAD. Additionally, the quality of the individual signals is set to POOR.

The cold leg temperature input signals from the dual-element RTD in each cold leg are also subjected to range and consistency checks, and then averaged to provide a group value for T(cold). If these signals agree within an acceptable interval (±DELTAC), the group quality is set to GOOD. If the signals do not agree within ±DELTAC, the group quality is set to BAD and the individual input signal qualities are set to POOR. One cold leg temperature input signal per loop may be deleted manually. The remaining T(cold) input signals will provide the loop T(cold) temperature signal.

For each loop, the TAS processes the T(hot,ave) and the T(cold) to produce a loop average temperature, T(avg), and a loop differential temperature (Delta T). The resulting signals are converted from digital to analog signals, then used to provide inputs to the plant computer, control system, and indication devices. The protection grade channels are isolated from the control systems with the same model of isolators used in the Sequoyah Eagle 21 system.

The hot leg temperature DELTAH criterion for each loop is an input parameter based upon temperature distribution tests within the hot leg, and is entered via a portable Man-Machine Interface (MMI). The cold leg temperature DELTAC criterion for each loop is an input parameter based upon operating experience, and is also entered via the MMI. One DELTAH and one DELTAC is required for each coolant loop.

A "Trouble" alarm and annunciator window common to all three loops will be added. This alarm is actuated when the T(hot,ave) value for a coolant loop is set to POOR. An "RTD Failure" alarm and annunciator is actuated when the T(cold) or T(hot,ave) group value for a coolant loop is set to BAD as described above. This alarm and annunciator informs the operator that there is an invalid T(cold) or T(hot,ave) group value for a loop. When the group quality is set to BAD, the channel status is changed to INOPERABLE, and the channel is placed in a tripped state.

The effect of the RTD channels on the Reactor Protection System will remain the same as previously utilized. For example, two-out-of-three voting logic channels continue to be utilized with the Model 7100 process control bistables continuing to operate on a "de-energize to actuate" principle.

The licensee requested a quarterly surveillance interval and a 4-hour period during which a channel may be placed in bypass for testing for the racks being upgraded with the Eagle-21 process protection equipment.

2.2 Analysis

The licensee presented information regarding the response time of the new RTD measurement system and also the accuracy of the new method for measuring the hot leg temperature by scoop mixing as designed by Westinghouse. The response time and accuracy affect the accident analyses.

2.2.1 RTD Response Time

As shown in the tabulation below, the response time for OTdT for the proposed system has some gains and losses compared to the existing RTD bypass system, but the total response time of the proposed system remains the same as for the existing system (6.0 sec).

RESPONSE TIME PARAMETERS FOR RCS TEMPERATURE MEASUREMENT

	RTD Bypass System	Fast Response Thermowell RTD System
RTD Bypass Piping and Thermal Lag (sec)	2.0	N/A
RTD Response Time (sec)	2.5	4.0
Electronics Delay (sec)	1.5	2.0
Total Response Time (sec)	6.0 sec	6.0 sec

The Technical Specification limit is 6.0 seconds.

The licensee states that the increase in the electronics time delay is conservative because the actual electronics time delay is significantly less then the 2.0 seconds claimed in the licensee's submittal. The staff finds the licensee's conservative value for electronics time delay to be acceptable.

NUREG-0809, "Resistance Temperature Detector Time Response Characteristics," points out that RTD response times have been known to degrade and that the loop current step response (LCSR) methodology is the recommended on-site method for checking RTD response times. Turkey Point Units 3 and 4 have upgraded custom Technical Specifications (TSs) which do not include a section for checking the RTD response time. However, the licensee has stated in a letter dated December 13, 1990, that they will perform RTD response time testing using the recommended LCSR method for checking the RTD response time. The on-site response time testing of the RTDs will be performed once every three refueling cycles (a refueling cycle is 18 months), on a staggered test basis, with one channel being tested each refueling cycle.

Based on the above information, the staff finds that the RTD response time has been addressed in an acceptable manner.

2.2.2 RTD Installation

The new method of measuring each hot leg temperature with three thermowell RTDs, used in place of the RTD bypass system with three scoops, has been analyzed to be at least as effective as the RTDs in the existing bypass system. The scoops are used to obtain a sampling of the flow at three 120-degree sectors in each of the hot legs in order to obtain a more accurate hot leg average temperature that accounts for the non-uniform temperature streaming. Formerly the RTD bypass system took the sampled flows from the scoops and made an external RTD temperature measurement in a plenum section. The new method with the RTD bypass system removed will measure coolant flow with a dual-element Weed RTD mounted in a thermowell. The Weed RTD is mounted to line up with the center hole of the five holes in the scoop. There are several locations where there is structural interference which prevents putting the new Weed RTDs in the scoops. In these cases new penetrations will be made and the Weed RTDs will be inserted to the same depth as those in the scoops, which is the center hole depth.

The RTD thermowells have been manufactured under Section III of the ASME Boiler and Pressure Vessel Code. Other piping modifications have been designed and fabricated consistent with the original design code, B31.1 (1955). Welding and inspections are consistent with Section XI of the ASME Code. Use of these codes in design, fabrication, and inspection of the RTD thermowells is appropriate and acceptable.

2.2.3 RTD Uncertainty and Calibration

The dual-element Weed RTD has improved accuracy over the existing RTDs. The total uncertainty includes a value for drift in addition to the normal accuracy (includes hysteresis and repeatability). This has been appropriately incorporated in the setpoint analysis.

The licensee committed, in a letter dated January 28, 1991, to obtain confirmatory information on the mixed mean temperature accuracy. This will be done by comparing pre-installation and post-installation calorimetric data from the RTD temperature measurements in the Turkey Point plants. The differences will be reconciled and the licensee will make this data available to the staff.

Because three RTDs are used to measure each hot leg temperature instead of the former single measurement, the error associated with the hot leg measurement is reduced to one over the square root of three compared to a single RTD. The uncertainty impact of the additional electronics needed for the two additional hot leg RTDs per loop has been evaluated by the licensee to be minimal.

The three signals are averaged to obtain the loop's T_{hot} value. The existing overall channel functional checks and calibration accuracy requirements are to be maintained. The impact of the rack drift has been considered in the evaluation.

The only change to the cold leg electronics is the averaging of two readings of the two RTDs in each leg. Therefore, there is no impact to the cold leg reading other than the increase in accuracy from the average of two RTD readings.

The net result of the proposed RTD bypass system modification is a slight improvement in the accuracy of the temperature-related functions over the accuracy now achievable with the existing RTDs in the bypass system. The licensee has reviewed the impact of the proposed modifications against the Turkey Point setpoint study to verify that the accuracy of the temperature-related functions are met.

A flow measurement uncertainty analysis, which considered the new RTD temperature measurement system, resulted in a calculated value of 3.4% (3.5% including a 0.1% fouling penalty). Turkey Point presently assumes a 3.5% uncertainty in primary flow determination, which will remain. The licensee stated in a letter dated December 13, 1990, that they will perform a cross-calibration of all RTDs during each refueling cycle by comparing the installed RTDs to each other. This will ensure the proper applicability of the temperature parameter as presented in the flow measurement uncertainty analysis.

By letter dated December 13, 1990, the licensee stated that RTD calibrations will consist of using the average of the RTD indicated temperatures as the reference temperature for cross-calibration purposes. The staff is concerned that using an average RTD signal as the reference signal for the RTDs, without an independent verification of the actual coolant temperature, may lead to a net drift of average value of the temperature indicated by the RTDs after several refueling outages. A licensee review of Weed RTD drift data indicated that the Weed RTD drift is random (does not consistently trend up or down) and is less than that assumed in the licensee's analyses (+1.2°F per refueling cycle). Based upon this conclusion, the licensee states that their calibration methodology is valid. The staff accepts the licensee's justification of this calibration methodology; however, the staff requires the licensee to maintain a historical record of the as-found and as-left calibration data and the surveillance data of each RTD to ensure that the cumulative drift of the RTDs remains within the safety analysis envelope.

2.2.4 RTD Failure Detection

A median signal selector (MSS) that satisfies the control and protection interaction requirements of IEEE 279-1971 will replace the existing high auctioneered T(avg) and delta T signals presently used in the control channels. There will be a separate MSS for each function. The MSS will use as inputs the protection grade T(avg) or delta T signals from all three loops, and will supply as an output the channel signal that is the median of the three signals. The effect will be that the various control grade systems will continue to use a valid RCS temperature in the case of a single signal failure. The staff has approved use of this MSS design for Beaver Valley Unit 2, and accepts the use of the MSS at Turkey Point Units 3 and 4.

To ensure proper action by the MSS, the existing manual switches for defeating a T(avg) or delta T signal from a single loop will be eliminated. The MSS will automatically select a valid signal in the case of a signal failure. Warnings that a loop signal failure has occurred will be provided by T(avg) and delta T deviation alarms. This method of detecting a defective RTD channel is acceptable.

The licensee will replace the existing Model 7100 process electronics monitoring RCS temperature for the overtemperature, overpower, T(avg) low-low, loss of flow, and pressurizer level with the Eagle 21 Process Protection System for each affected protection set. The licensee will remove all existing 7100 modules for these channels and use the modules as spares in other protection channels. The two-out-of-three voting logic will remain the same. The Eagle-21 TAS is identical to the TAS used in the Sequoyah Nuclear Plant, and was reviewed by the staff. The licensee further states that the isolation devices are the same as those used at Sequoyah, and found acceptable by the staff. The staff's review of the Sequoyah Eagle-21 system found the TAS (as part of that system) to be acceptable. Since the TAS design with isolation devices proposed for the Turkey Point plants is identical to the TAS design used at the Sequoyah plant and approved by the staff, the staff accepts the licensee's implementation of the Eagle-21 TAS at Turkey Point Units 3 and 4.

For testing of the racks being upgraded with the Eagle-21 process protection equipment, the licensee requests a quarterly surveillance interval and a 4-hour period during which a channel may be placed in bypass. These times are consistent with the intervals previously approved by the staff in its review of Westinghouse Topical Report WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and in its review of EAGLE-21 process protection system implementation at Sequoyah Nuclear Plant, as reflected in the Sequoyah Safety Evaluation Report dated May 16, 1990.

2.2.5 Non-LOCA Accidents

The impact of the RTD bypass elimination for Turkey Point on FSAR Chapter 15 non-LOCA accidents has been evaluated by the licensee. Since the effect of the temperature response time and accuracy of the new system is not degraded, the former conclusions in the FSAR remain valid.

2.2.6 LOCA Evaluation

The elimination of the RTD bypass system has been found not to impact the uncertainties associated with RCS temperature and flow measurement. It is concluded therefore that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint without requiring any detailed reanalysis.

2.2.7 f(delta-I) Penalty

FPL proposed to eliminate the f(delta-I) penalty function for the OPdT setpoint and reduce the slope of the f(delta-I) penalty function for the OTdT setpoint.

The f(delta-I) penalty function for the OPdT setpoint was previously used for the OPdT trip setpoint. This trip protects against excessive linear heat generation rates which could cause high fuel centerline temperatures. The elimination of the f(delta-I) penalty function for the OPdT setpoint is the result of approved Westinghouse technical setpoint methodology. Also, the current standard Westinghouse Technical Specifications no longer include the f(delta-I) penalty function for the OPdT setpoint. Therefore, we find this change to be acceptable.

The f(delta-I) penalty function is also used for the OTdT trip setpoint. This trip protects the core against departure from nucleate boiling (DNB). The requested reduction of the f(delta-I) slope in the penalty function for the OTdT setpoint is in the non-conservative direction. However, the amount of reduction requested falls in the acceptable range based on an approved Westinghouse setpoint methodology, and will provide adequate DNB protection. Therefore, we find the change to be acceptable.

2.3 Summary of Technical Specifications Changes

As a result of the modifications associated with the removal of the RTD bypass manifolds, and the installation of the Weed RTDs and the Eagle-21 TAS, the licensee proposed several amendments to the Turkey Point Units 3 and 4 Technical Specifications. The staff reviewed the following changes.

Change 1 Table 2.2-1 will be changed to the five-column format, consistent with the Westinghouse STS format. The new format does not change the plant safety margins, and is therefore acceptable.

Change 2 In Table 4.3-1, pages 3/4 3-8 and 3/4 3-12, references to the RTD bypass loops were removed. These changes are acceptable.

Change 3 On page 2-4, Reactor Coolant Flow, an allowable value of 88.7% was added. This change is acceptable because the margin between the total allowance and the channel statistical allowance is greater than the margin between the trip setpoint of 90% and the allowable value.

Change 4 In Tables 4.3-1 and 4.3-2, the analog channel operational test surveillance interval was changed from monthly to quarterly. This change is consistent with the staff's prior acceptance of Westinghouse Topical Report WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System."

Change 5 In Tables 3.3-1 and 3.3-2, pages 3/4 3-2, 3/4 3-7, 3/4 3-15, 3/4 3-18, 3/4 3-22, the digital channel operational test surveillance interval is changed from monthly to quarterly. This change is consistent with the staff's prior acceptance of Westinghouse Topical Report WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and subsequent evaluation for digital process equipment.

Change 6 In Table 2.2-1, page 2-4, a Pressurizer Water Level High allowable value of 92.2% is specified. This change is acceptable because the margin between the total allowance and the channel statistical allowance is equal to the margin between the trip setpoint of 92% and the allowable value.

Change 7 In Table 2.2-1, page 2-7, the OTdT function is standardized and accounts for the removal of the RTD bypass lines. This change is acceptable because the revision correctly accounts for the elimination of the direct immersion RTDs and the RTD bypass lines.

Change 8 In Table 2.2-1, page 2-8, the f(delta-I) gain is reduced to 1.5, and an allowable value of 1.5% is added. This change is acceptable based upon Westinghouse safety evaluation results, and its application in the Westinghouse Setpoint Methodology.

<u>Change 9</u> In Table 2.2-1, page 2-10, the f(delta-I) gain is removed, and an allowable value of 1.4% is added. This change is acceptable based upon Westinghouse safety evaluation results, and its application in the Westinghouse Setpoint Methodology.

<u>Change 10</u> In the Bases section, page B 2-5, the reference of axial power distribution from the bases for the OPdT is removed. This is an editorial change that accounts for the removal of the f(delta-I) gain from the OPdT function.

<u>Change 11</u> In Table 3.3-3, the T(avg) trip setpoint increases from 531°F to 543°F, and an allowable value of 542.5°F is added. The format was also changed to a six column format. The increased trip setpoint temperature is acceptable because it is more conservative than the existing trip setpoint. Although the allowable value of T(avg) is less than the trip setpoint value, this change is acceptable because the margin between the total allowance and the channel statistical allowance is greater than the margin between the trip setpoint and the allowable value.

<u>Change 12</u> In Definitions, page 1-6, a definition for Digital Channel Operational Test is added. The definition is acceptable.

<u>Change 13</u> In Section 2.2, Limiting Safety System Settings, page 2-3, a qualifier to account for calibration tolerance is added and defines the methodology for evaluating Equation 2.2-1. This change is consistent with the application of the Westinghouse Setpoint Methodology, and is acceptable.

<u>Change 14</u> In Table 2.2-1, page 2-4, Items 5 and 9, the values of Z + S exceed the total allowance, but are acceptable because the Action statements applying Equation 2.2-1 require the licensee to apply "as measured" values for the sensor when the value of S is such that Equation 2.2-1 is not satisfied. If the "as measured" value for S is such that Equation 2.2-1 cannot be satisfied, the affected channels are declared INOPERABLE until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the trip setpoint value. Since the value of S in Items 2 and 9 of Table 2.2-1 are based upon very conservative estimates of sensor uncertainties, use of the "as measured" sensor uncertainties is appropriate for this application.

<u>Change 15</u> In Section 3/4.3.3, page 3/4 3-13, a qualifier to account for calibration tolerance is added and defines the methodology for evaluating Equation 2.2-1. This change is consistent with the application of the Westinghouse Setpoint Methodology, and is acceptable.

3.0 SUMMARY

The staff concludes that the modified RTD system is not functionally different from the existing system except for the use of three RTDs instead of one in each hot leg. The RTS and ESFAS will operate as before. Section 7 of the FSAR remains valid. The additional electronics for averaging the three hot leg RTD signals and the associated isolation devices are the same as those approved by the staff for use in the Sequoyah Nuclear Plant, and are consequently acceptable for use in the replacement of the existing RTD Bypass system with the proposed temperature measurement system. The MSS is the same as the design approved by the staff for the Beaver Valley, Unit 2, and is also acceptable.

The licensee's request for a quarterly surveillance interval and a 4-hour period during which a channel may be placed in bypass for testing for the racks being upgraded with the Eagle-21 process protection equipment is consistent with the intervals previously approved by the staff in its review of Westinghouse Topical Report WCAP-10271-P-A, and in its review of EAGLE-21 process protection system implementation at Sequoyah Nuclear Plant.

The staff requires the licensee to maintain historical records of the as-found and as-left calibration and surveillance data for each RTD. These records will ensure cumulative RTD drift remains within the safety analysis envelope.

To support the modifications required to eliminate the RTD bypass manifold system, the licensee proposed changes to the Turkey Point plant TS. The TS revisions are a result of changing the Turkey Point TS format to a 5-column standard format, and specific revisions to address changes in instrumentation and system uncertainties. Evaluations performed by the licensee and reviewed by the staff show the uncertainty values to be acceptable. Technical specification changes revise the use of the power imbalance function, and are acceptable to the staff. The licensee increased the Tavg-LOW trip setpoint to a more conservative value based upon the results of the vendor's setpoint analyses. This change is also acceptable to the staff. Additionally, the licensee proposed changes in the operational test surveillance intervals and the allowable outage times for INOPERABLE channels. These changes are acceptable.

4.0 STATE CONSULTATION

Based upon the written notice of the proposed amendments, the Florida State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 891). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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