

TECHNICAL SPECIFICATION CHANGE REQUEST NO. 94

Replace Appendix A pages 2-6, B2-7 and 3/4 3-6 with the attached revised pages 2-6, B 2-7 and 3/4 3-6.

Reason for Proposed Changes

- A. Pages 2-6 and B 2-7 - Florida Power Corporation agreed to take credit for the pump 120% overpower trip to resolve NRC concern that faulted conditions in the Reactor Coolant Pump Power Monitors (RCPPM) could mask loss of flow conditions. These changes address those concerns.
- B. Page 3/4 3-6 - The change to Item 8 is proposed to allow the installation of additional time delay relays in the RCPPM circuitry to prevent spurious trips.

Furthermore, this change corrects a previous error supplied by Florida Power Corporation based on a misunderstanding of Babcock and Wilcox supplied data. Technical Specifications Change Request No. 77 proposed an RCPPM response time of 0.470 seconds which was based on BAW - 1684 "Crystal River Unit 3 Cycle and Reload Report". This total was comprised of the following components:

.240	Sensor (CT, PT, WT, etc.)
.150	Reactor Protective System
.080	CRD Breaker Delay
<u>0.470</u>	TOTAL

This was that set of equipment that Babcock and Wilcox felt could and should be tested in place. This was never utilized as an analytical assumption, nor did it reflect corrections due to the "RPS Instrument Inaccuracy Concern." Two other changes to this page were part of this error re-analysis but not the RCPPM value.

The changes to the other channels (Items 2, 5 and 6) are proposed to resolve a difference between the definition of response time between the FSAR and Technical Specifications. The FSAR analyses include as response time, the time between the monitored parameter change and rod insertion. The Technical Specifications only include change in parameter through opening of the control rod drive breakers. The remainder of the trip string (CRDM release delay and rod insertion are addressed by Technical Specification 3.1.3.4, (ROD DROP TIME). This difference of 0.060 seconds was not correctly reflected in the Technical Specification tables.

Safety Analysis

The safety analyses used the following information for response time.

TABLE 1

<u>Functional Unit</u>	<u>Sensor & RPS Delay</u>	<u>Breaker Delay</u>	<u>CRD Release Delay</u>	<u>Dedicated Margin for Uncertainty</u>	<u>Delay Time Assumed for FSAR Anal.</u>
Nuclear Overpower	0.186	0.080	0.060	0.000	0.326
Nuclear Overpower based on RCS flow and Axial Power Imbalance	1.690	0.080	0.060	0.020	1.850
RCS Pressure High and Low	0.261	0.080	0.060	0.099	0.500
Pump Status Based on (RCPPM's)	0.480	0.080	0.060	0.000	0.620

Using the Technical Specification definition, response times assumed in the FSAR supporting analyses should be decreased by .060 seconds to the values on the attached revised Table 3.3-2. Operation with these time delays is, therefore, supported by the safety analyses and will not constitute an unreviewed safety question.

TABLE 2
EVALUATION OF RCPPM RESPONSE TIMES

<u>COMPONENT</u>	<u>ORIGINAL DESIGN RESPONSE TIME</u>	<u>POST CHANGE ADDING OPTIC ISOLATORS</u>	<u>POST CHANGE ADDING TIME DELAY RELAYS</u>	<u>CURRENT RESPONSE TIME</u>
CT/PT	Included w/Watts Transducer	16 ms	16 ms (1)	20 ms (2)
Watts Transducer (3)	240 ms (4)	72 ms (5)	72 ms	72 ms
Time Delay Relay	150 ms (4)	139 ms (6)
Optic isolator	150 ms (4)	106 ms (9)	106 ms
RPS	150 ms (4)	131 ms (7)	131 ms	} 104 ms (8)
CRD Breaker	80 ms (4)	80 ms	80 ms	
Subtotal	470 ms	451 ms	555 ms	441 ms
Margin	90 ms	109 ms	5 ms	119 ms
Total	560 ms	560 ms	560 ms	560 ms
Release Delay	60 ms (4)	60 ms	60 ms	60 ms
Safety Analysis Assumption	620 ms	620 ms	620 ms	620 ms

Allocated 240 ms. Results in 2 ms. margin

NOTES

- (1) GE Documentated Analysis Results Available
- (2) Allocated Response Time (G.E. Documentated Analysis Results Available)
- (3) Includes Watts Transducer, Bistable and Output Relay.
- (4) Allocated Response Time
- (5) B&W Test (By Rochester). Recent Test Date Indicate Actual Response Time of 22 ms (PT only). 72 ms. Represents Worst Case Including: Watts Transducer, Bistable and Clare Output Relay.
- (6) Tested at FPC. Maximum Delay on Any Relay Was 139 ms. Each Relay Tested Five Times.
- (7) Specification Requirements Supported by Type Test Data (See BAW - 10003 and/or references therein).
- (8) SP-115 Step 6.5.1.1 Includes Entire Channel From Output of Flux Monitor through CRD Breaker Opening, (i.e., RPS & CRD Breaker & Additional Components). 104 ms. is Worst Case. Test is Periodically Repeated.
- (9) Type Test Data at 65°F at 95% Confidence Level. Testing at Ambient Temperatures (74°F) Yielded Even Lower Values.

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTION UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
8. Pump Status Based on Reactor Coolant Pump Power Monitors (1)	More than one pump drawing ≤ 3900 KW or ≥ 9000 KW	More than one pump drawing ≤ 3900 KW or ≥ 9000 KW
9. Reactor Containment Vessel Pressure High	≤ 4 psig	≤ 4 psig

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- (1) Trip may be manually bypassed when RCS pressure ≤ 1720 psig by actuating Shutdown Bypass provided that:
- a. The Nuclear Overpower Trip Setpoint is $\leq 5\%$ of RATED THERMAL POWER
 - b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤ 1720 psig is imposed, and
 - c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint ≤ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure - Low trip.

Reactor Coolant Pump Power Monitors

In conjunction with the power/imbalance/flow trips, the Reactor Coolant Pump Power Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to more than one reactor coolant pump not operating.

A reactor coolant pump is considered to be not operating when the power required by the pump is $\geq 120\%$ or is $\leq 70\%$ of the nominal operating power. The nominal operating power decreases from when a pump is first started during heatup and is pumping dense fluid (typically 7500 KW) to when a pump is operating at full reactor power and is pumping less dense fluid (typically 5500KW). In order to avoid spurious trips during normal operation, the 120% trip setpoint (9000KW) is based on the nominal operating power for a pump during heatup and the 70% trip setpoint (3900KW) is based on the nominal operating power for a pump operating at full reactor power. Florida Power has agreed to take credit for the pump overpower trip in order to assure that certain potential faults such as a seismically induced fault high signal will not prevent this instrumentation from providing the protective action (i.e., a trip signal).

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

<u>Functional Unit</u>	<u>Response Times</u>
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower *	\leq 0.266 seconds
3. RCS Outlet Temperature - High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE *	\leq 1.79 seconds
5. RCS Pressure - Low	\leq 0.44 seconds
6. RCS Pressure - High	\leq 0.44 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Pump Status Based on RCPs	\leq 0.56 seconds
9. Reactor Containment Pressure - High	Not Applicable

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.