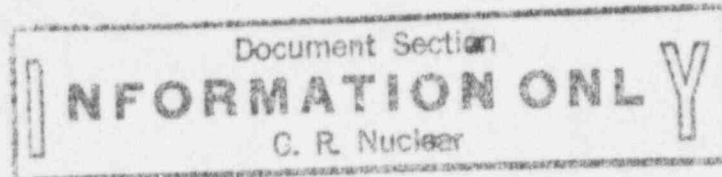


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SURVEILLANCE PROCEDURE

SP-406

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

REFUELING OPERATIONS DAILY DATA REQUIREMENTS

THIS PROCEDURE ADDRESSES SAFETY RELATED COMPONENTS

APPROVED BY: Interpretation Contact

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(SIGNATURE ON FILE)

DATE: 3/1/96

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1.0 PURPOSE

The purpose of this procedure is to provide a checklist of Technical Specification requirements that must be met prior to and during Mode 6, and when Refueling Operations are in progress.

2.0 REFERENCES

2.1 IMPLEMENTING REFERENCES

None

2.2 DEVELOPMENTAL REFERENCES

2.2.1 Technical Specification References

<u>Applicable References</u>	<u>Surv. Perf. During Modes</u>	<u>LCO/Other Requirements During Modes</u>	<u>Surv. Freq.</u>	<u>Freq. Notes</u>	<u>Mode Notes</u>
3.9.1.1	5,6	6	SP-1	39	
3.9.1.1	6	6	SP-2		29
3.9.1.1	5,6	6	SP-3	39	
3.9.1.1	5,6	6	SP-4		
3.9.2.1	5,6	6	S	39	
3.9.4.1	5,6	6	S	39	47
3.9.5.1	6	6	S	39	48
3.9.6.1	6	6	D	14	29

SURVEILLANCE FREQUENCY:

S - At least once per 12 hrs.

D - At least once per 24 hrs.

Q - At least once per 92 days.

SP-1 - Prior to removing or unbolting the Reactor Vessel head.

SP-2 - Prior to the withdrawal of the first safety or regulating rod (Groups 1 thru 7) in excess of 3 ft. from its fully inserted position.

SP-3 - At least once per 72 hours.

SP-4 - Special frequency or special case when analyses are inadequate.

FREQUENCY NOTES:

- 14 - Within 2 hrs. prior to movement of fuel assemblies or control rods.
- 39 - Establish surveillance prior to descent into applicable mode.

MODE NOTES:

- 29 - During core alterations.
- 47 - With the water level \geq 156 ft.
- 48 - With the water level $<$ 156 ft.

2.2.2 FSAR 9.6.2.3

3.0 PERSONNEL INDOCTRINATION

3.1 SETPOINTS

None

3.2 DESCRIPTION

- 3.2.1 This procedure provides a means for monitoring specific plant conditions and equipment availability prior to and during Refueling Operations as required by Technical Specifications.

3.3 DEFINITIONS

3.3.1 Core Alteration

CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

3.4 RESPONSIBILITIES

- 3.4.1 The CR-3 Operations section shall be responsible for the content of this procedure.
- 3.4.2 The CR-3 Nuclear Shift Supervisor on Duty shall ensure that this procedure has been completed as specified by Step 2.2.1.
- 3.4.3 The CR-3 Reactor Engineer shall be responsible for providing Operations with the most restrictive boron concentration as specified in Step 3.7.2.
- 3.4.4 This procedure is designed and written to be performed by Nuclear Operators reporting directly to the Shift Supervisor. There are no additional skills required.

3.5 LIMITS AND PRECAUTIONS

- 3.5.1 IF at any time the requirements of Enclosure 1 cannot be met, THEN immediately refer to the applicable Technical Specifications (TS) action statement in the TS Action Column.
- 3.5.2 For work located in Radiation Controlled Areas, due consideration must be given to the ALARA program. This may result in a determination that special preparations and/or precautions are necessary.
- 3.5.3 Boron concentration analysis that deviate ± 25 ppm from the last analysis should be reverified.

3.6 ACCEPTANCE CRITERIA

- 3.6.1 The data recorded on Enclosure 1 is within the tolerance listed under the "Required" column.

3.7 PREREQUISITES

- 3.7.1 A DHR loop shall be determined to be operating and circulating RC at a flow rate ≥ 2700 gpm at least once every 12 hours.
- _____
Initial/Date
- 3.7.2 Boron concentration is per the Core Operating Limits Report. Concur with Reactor Engineering.

- 3.7.3 Determine that the boron concentration analyses meet the Acceptance Criteria established by Step 3.7.2.

Initial/Date

- 3.7.4 Notify the Nuclear Shift Supervisor on Duty prior to the performance of this procedure.

Initial/Date

- 3.7.5 Personnel Indoctrination section has been read and understood.

Initial/Date

4.0 INSTRUCTIONS

4.1 PRIOR TO ENTERING MODE 6

- 4.1.1 Complete Section "A" of Enclosure 1 prior to entering Mode 6.

4.2 DURING MODE 6

- 4.2.1 Perform Section "A" of Enclosure 1 once per shift while in Mode 6.

4.3 PRIOR TO CORE ALTERATIONS

- 4.3.1 Complete Section "B" of Enclosure 1 within ONE hour prior to CORE ALTERATIONS.

- 4.3.2 Verify that the Refueling Canal water level is greater than the 157 ft. elevation within TWO hours prior to FUEL or Control Rod movement.

- 4.3.3 Complete Section "C" of Enclosure 1 prior to FUEL or Control Rod movement within the Reactor Vessel.

4.4 DURING CORE ALTERATIONS OTHER THAN FUEL OR CONTROL ROD MOVEMENT

- 4.4.1 Perform Sections "A" and "B" of Enclosure 1 once per shift during CORE ALTERATIONS.

4.5 DURING FUEL OR CONTROL ROD MOVEMENT WITHIN THE REACTOR VESSEL

- 4.5.1 Perform Sections "A", "B", and "C" of Enclosure 1 once per shift during FUEL or Control Rod movements within the Reactor Vessel.

4.6 DEFUELED

- 4.6.1 Shift readings, per this procedure, may be suspended when the Reactor is defueled and no fuel elements are in the fuel transfer canal.

5.0 FOLLOW-UP ACTIONS

5.1 RESTORATION INSTRUCTIONS

None

5.2 CONTINGENCIES

- 5.2.1 Refer to Technical Specification Actions under the Action column on Enclosure 1.

SECTION A PRIOR TO MODE 6 AND DURING MODE 6

TS ACTION	DESCRIPTION	REQUIRED	00-08	08-16	16-24
			ACTUAL	ACTUAL	ACTUAL
3.9.2.1	NI-1 Source Range Test Module "On Test" Lamp	Dim (✓)			
3.9.2.1	NI-2 Source Range Test Module "On Test" Lamp	Dim (✓)			
3.9.2.1	NI-1-NI Flux Indication at Control Console	> 0.4 cps			
3.9.2.1	NI-2-NI Flux Indication at Control Console	> 0.4 cps			
3.9.2.1	NI-14-NI Flux Indication at Control Console	> 10 ¹ cps****			
3.9.2.1	NI-15-NI Flux Indication at Control Console	> 10 ¹ cps****			
3.9.2.1	NI-1-DNI Rate Indication at Control Console	-0.1 to +0.1 dpm			
3.9.2.1	NI-2-DNI Rate Indication at Control Console	-0.1 to +0.1 dpm			
3.9.2.1	Source Range Counts Audible in Containment	(✓)			
3.9.2.1	Source Range Counts Audible in Control Room	(✓)			
3.9.4.1: IOC NL94-0039	Decay Heat Flow A or B and Circulating RC	≥ 2700 gpm**			
3.9.4.1: IOC NL94-0039	Decay Heat Flow A or B and Circulating RC	≥ 500 gpm			
3.9.4.1: IOC NL94-0039	Reactor Coolant System Temperature (DHP Suction Temperature)	≤ 140°F			
3.9.5.1	Decay Heat Train A AND B operable OR Refueling Water Level ≥ 156' elev	(✓)			
3.9.1.1	Top of Refueling Canal (RB) Boron	≥ ***		N/A	N/A
3.9.1.1	Reactor Vessel Boron	≥ ***		N/A	N/A

SECTION B WITHIN 1 HOUR PRIOR TO CORE ALTERATION

	DESCRIPTION	REQUIRED	00-08	08-16	16-24
			ACTUAL	ACTUAL	ACTUAL
FSAR 9.6.2.4	Main Control Room - Refueling Stations Communications Check	(✓)			

SECTION C PRIOR TO AND DURING MOVEMENT OF FUEL OR CONTROL RODS

TS ACTION	DESCRIPTION	REQUIRED	00-08	08-16	16-24
			ACTUAL	ACTUAL	ACTUAL
FSAR 9.6.2.3 3.9.6.1	Refueling Water Level	> 157' elev. (✓)			
	Spent Fuel Pool Boron	***		N/A	N/A
FSAR 9.6.2.4	Last Criticality	> 72 hours (✓)		N/A	N/A

PERFORMED BY:

- NOTES: * Establish surveillance within 2 hours prior to movement. 23 ft. above fuel in the Vessel is equal to 148' Elevation; 23 ft. above fuel in the Spent Fuel Pool is 156' Elevation; 8 ft. above active fuel level of a raised assembly is 157' elevation which is more restrictive.
- ** Demonstrate flow > 2700 gpm once per 12 hours when water level ≥ 156 ft.
- *** Boron Concentration to be determined by Reactor Engineer and as per the COLR.
- **** IF using the Gamma-Metrics System (NI-14-NI or NI-15-NI) to take the place of the BF3 System (NI-1-NI or NI-2-NI), THEN the BF3 System is expected to indicate 10 to 30 times greater than the Gamma-Metrics System.
- IF not, contact Systems Engineering (Reference: Technical Support letter NPTS 96-0107).