Florida Power

CORPORATION

Crystal River Unit 3
Docket No. 50-302

December 3, 1997
3F1297-27

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Technical Specification Change Request Notice 210, Request for Additional Information (TAC No. M98991)

References:

1. FPC letter dated June 14, 1997 (3F0697-10) "Technical Specification Change Request Notice 210"
2. NRC letter dated November 26, 1997 (3F1197-23), "Crystal River Unit 3 - Request for Additional Information - License Amendment Related to Technical Specification Change Request No. 210, Small-Break Loss-of-Coolant-Accident (SBLOCA) Submittal"

Dear Sir:

In Reference 1, Florida Power Corporation (FPC) submitted Technical Specification Change Request Notice (TSCRN) 210, which proposes amendments to Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). TSCRN 210 is necessary to address design and licensing basis changes primarily involving plant systems used to mitigate the consequences of certain small break loss of coolant accidents (SBLOCA). In Reference 2, the NRC provided FPC with a request for additional information (RAI). FPC's response to the RAI is provided in Attachment A.

FPC suggests that a meeting be held December 10, 1997, to facilitate NRC review of FPC's responses to the RAI. During this meeting, FPC anticipates presenting the CR-3 Probabilistic Safety Assessment modeling discussed in Attachment A.

A001/1

There are no new commitments made in this submittal.

If you have any questions concerning this submittal, please contact Mr. David Kunsemiller, Manager, Nuclear Licensing at (352) 563-4566.

Sincerely,

John J. Holden
Director
Site Nuclear Operations

JJH/mal

cc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

Attachments:

A. Response to Request for Additional Information
B. EOP-13, Rules
C. EOP-14, Enclosure 17, Control Complex Emergency Ventilation
D. Instructional Outlines, ROT-9-200 & ROT-9-200A
E. AI-402C, AP and EOP Verification and Validation Plan

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT A

RESPONSE TO NRC RAI

PREFACE

NRC provided FPC with a request for additional information (RAI) in letter dated November 26, 1997 (3N1197-23). The NRC's RAI Requests 1, 2, and 3 ask for specific information regarding risk analysis associated with "Loss of Coolant Accidents (LOCA)/Loss of Offsite Power (LOOP)" events.

The subject of the RAI is FPC's Technical Specification Change Request Notice 210, dated June 14, 1997 (3F0697-10). The scope of TSCRN 210 requests certain license and design basis changes related to small break LOCAs. Consequently, FPC's responses to Requests 1, 2, and 3 address small break LOCAs. The CR-3 baseline Probabilistic Safety Assessment (PSA) addresses the risk contributions associated with other CR-3 accidents, in addition to small break LOCAs.

ATTACHMENT A
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION CHANGE REQUEST NOTICE 210

NRC REQUEST 1

Provide the initiating event (IE) frequency of Loss-of-Coolant Accident (LOCA)/Loss of Offsite Power (LOOP). Please provide the initiating frequency of a LOCA, the dependent or conditional probability of LOOP (i.e., the probability of a LOOP given that a LOCA has occurred), and the bases for these frequencies.

In a LOCA/LOOP accident scenario, as postulated in Generic Safety Issue (GSI) 171, "ESF [engineered safety features] failure from LOOP subsequent to LOCA, "there is an increase in the likelihood of a LOOP given a LOCA compared to a random (independent) occurrence of the LOOP in the same period. This increased likelihood can be due to a disturbance in the grid caused by the reactor trip which occurs after a LOCA, problems due to bus transfer, or due to the increased loads on the emergency busses in response to a LOCA." To address the issues raised as part of GSI 171, NUREG/CR-6538, "Evaluation of LOCA With Delayed LOOP and LOOP With Delayed LOCA Accident Scenarios" was published in July 1997. This report, in part, quantitatively analyzes LOCA/LOOP accident sequences.

FPC RESPONSE

FPC has completed a quantitative CR-3 risk analysis model associated with design and licensing changes proposed by TSCRN 210. The initiating event frequency for a small break LOCA/LOOP accident scenario was calculated in the CR-3 risk analysis as 2.24×10^{-5} per year. The initiating event frequency is based on a combination of the frequency of a Small Break LOCA (SBLOCA) and the conditional probability of a LOOP given a LOCA as discussed below.

Conditional Probability of a LOOP Given a LOCA

NUREG/CR-6538, "Evaluation of LOCA with Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios," estimated the conditional probability of a LOOP given a LOCA as 1.4×10^{-2} per year. The CR-3 risk analysis addressing TSCRN 210 assumes that the LOCA and LOOP are not independent and uses the same frequency for a conditional LOOP given a LOCA as NUREG/CR-6538.

Frequency of a SBLOCA

The small break LOCA frequency used in the CR-3 Probabilistic Safety Assessment (PSA) model is 1.6×10^{-3} per year and is based on industry data up to December 31, 1996. There have been only two small break LOCAs in the U.S. nuclear industry: an RCP seal failure at Arkansas Nuclear One in 1980, and an instrumentation line failure at Oconee on November 25, 1991. The second of these events is included in the SBLOCA frequency for PWRs of 3.76×10^{-4} per year given in EPRI report TR-102266, "Pipe Failure Study Update." The number of U.S. PWR reactor years through December 31, 1996, is estimated to be 837.8 years based on extrapolated U.S. power reactor performance data obtained from the American Nuclear Society. Therefore, the CR-3 PSA model estimate for the frequency of a SBLOCA for U.S. PWRs is calculated as:

$$f\text{sbloca} = 1 \text{ event}/837.8 \text{ reactor-years} + 3.76 \times 10^{-4} \text{ per reactor-year}$$

$$= 1.6 \times 10^{-3} \text{ per reactor-year}$$

NUREG/CR-6538, Table 4.3 identifies the frequencies of a SBLOCA as 1×10^{-3} based on NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." The CR-3 PSA estimate for the frequency of a SBLOCA is 60% higher than the frequency used in the NUREGs and, therefore, is a conservative estimate.

Initiating Event Frequency

Combining the frequency of the small break LOCA with the conditional probability of a LOOP given a LOCA, the CR-3 risk analysis estimate for the frequency of a small break LOCA and a LOOP is calculated as:

$$f\text{sbloca/loop} = 1.6 \times 10^{-3} \text{ per reactor-year} \times 1.4 \times 10^{-2} \text{ per year}$$

$$= 2.24 \times 10^{-5} \text{ per year.}$$

NUREG/CR-6538, Table 4.3 identifies the frequency of a SBLOCA with a conditional probability of a LOOP given a LOCA as 1.4×10^{-5}. Because of the conservative estimate for the frequency of a SBLOCA, the CR-3 risk analysis model estimate for the frequency of such an initiating event is also 60% higher than that used in NUREG/CR-6538.

NRC REQUEST 2

Given a LOCA/LOOP initiating event, what are the plant's mitigating actions, including automatic/manual system/equipment response and operator actions? Please account for all plant and procedural changes (including operator actions and new load management strategy). What are the assigned failure, unavailability, and human error probabilities associated with these mitigating actions?

FPC RESPONSE

The CR-3 actions to mitigate the SBLOCA scenarios are summarized in TSCRN 210, Attachment B, "Safety Assessment." The design and licensing basis changes addressed by TSCRN 210 include new operator actions and modifications involving automatic equipment.

To assist in the review, the CR-3 modifications involving automatic equipment for TSCRN 210 are presented in Table 1. Table 1 explains the purpose of each of these modifications and how risk modeling was addressed.

The new operator actions proposed by TSCRN 210, including those actions associated with the load management strategy, are identified by the enclosed Table 2. The new operator actions identified in Table 2 are those that were added to the Emergency Operating Procedures as a result of TSCRN 210. The other operator actions associated with TSCRN 210 that have not been previously reviewed by the NRC already existed in CR-3 procedures prior to TSCRN 210 and are not considered to be new operator actions.

Each of these operator actions of Table 2 are numbered to correspond to the operator actions identified in Table 3B of FPC's letter dated September 25, 1997 (3F0997-30). For each of these new operator actions, Table 2 identifies the associated failure scenario described in the Safety Assessment, the basis for the operator action, the risk modeling used for the operator action, and the assumed human error probability. The CR-3 risk analysis model used an assumed human error probability of 1.0 for each operator action, as noted in Table 2, except for the periodic re-evaluation of the HPI line break criteria on RCS repressurization (OA # 17).

A human error probability of 1.0 assumes that the operator fails to perform the required task 100% of the time. FPC considers a human error probability of 1.0 to be extremely conservative, since each operator action identified in Table 2 is addressed by explicit procedure guidance, has been included in operator training, and can be consistently completed within the required timeframes as shown by recent simulator exercises.

The operator action associated with the periodic re-evaluation of the HPI line break criteria (OA #17) was not included in the CR-3 risk analysis model since the frequency of a SBLOCA/LOOP with the requirement that the SBLOCA occur in an HPI line is already very small, approximately 7×10^{-7} per year. Such a failure of the operator to complete the required action would not have an appreciable effect on the frequency of core damage.

NRC REQUEST 3

Based on the above parameter estimates, what is the calculated core damage frequency contribution from the LOCA/LOOP sequences for the proposed changes?

FPC RESPONSE

The CR-3 risk analysis calculated a baseline core damage frequency due to a SBLOCA to be 3.25×10^{-6} per year when taking into account SBLOCA/LOOP dependency. The baseline calculation was then modified to reflect the changes due to these new operator actions as described in Table 2 and the conditional probability of a LOOP given a SBLOCA. As discussed in FPC's responses to NRC Requests 1 and 2, the probabilities of human error associated with these new operator actions were conservatively assumed to be 1.0, and the conditional probability of a LOOP given a SBLOCA was assumed to be 1.4×10^{-2} per year consistent with NUREG/CR-6538. The SBLOCA sequences were requantified, and their core damage frequency contribution was calculated to be 3.57×10^{-6} per year. This is an absolute increase in core damage frequency of 3.2×10^{-7} per year.

The current core damage frequency of CR-3 for all internal events is 7.13×10^{-6} per year. Taking into account SBLOCA/LOOP dependency, the core damage frequency is 7.19×10^{-6} per year. The core damage frequency from all internal events, when conservatively assuming human error probabilities of 1.0 for the new operator actions proposed by TSCRN 210, is 7.51×10^{-6} per year.

In summary, the contribution to core damage frequency of the new operator actions proposed by TSCRN 210 has been shown to be minimal even when assuming conservative human error probabilities of 1.0. FPC considers that this analysis demonstrates the changes in procedures and the new load management strategies proposed by TSCRN 210 do not have an appreciable effect on the low risk of core damage frequency at CR-3.

NRC REQUEST 4

Validation and verification (V/V, should be conducted for operator actions associated with each of the three single failure events. If either Loss of Battery A (LOBA) or Loss of Battery B (LOBB) is not tested, justification should be provided.

FPC RESPONSE

Verification and Validation (V&V) as described in NUREG 1358 Supplement 1 is intended to back up the use of complete and accurate control documents in the development and revision of the procedures. It is an integral part of the procedure development to assure the procedures are correct before they are implemented. FPC implements the provisions of NUREG 1358 via Administrative Instruction AI-402C, "AP and EOP Verification and Validation Plan," a copy of which is attached to this letter.

The attached Table 3 identifies the simulator validations covering each of the operator actions. Recorded times for these actions were provided to the NRC for staff review in FPC letters dated September 25, 1997 and November 19, 1997. The validations were performed using a combination of minimum and full shift complements.

For the actions associated with small break LOCA mitigation, the proposed final versions of the procedures are not significantly different than the draft procedures which were validated. This is confirmed by a composite evaluation of each step and associated flow paths performed as part of the final review process. As can be seen from the attached list, two operator actions were not validated on the simulator for the three failure scenarios discussed in TSCRN 210. For Operator Action # 8, transfer borated water storage tank (BWST) suction to the reactor building sump, the need for this transfer did not manifest itself in a timely manner to accommodate the simulator runs performed for the small break LOCA scenarios that pertain to TSCRN 210 (for the small break sizes in question, it would take many hours to reach the required swapover level). However, other simulator scenarios using the same operator actions were performed during which the accuracy and completeness of EOP-3, "Inadequate Subcooling Margin," Step 3.13 was demonstrated. These included a table top validation performed on November 10, 1997 to validate a loss of subcooling margin (SCM) with no high pressure injection (HPI) and a simulator validation performed on September 19, 1997 for a HPI cold leg break with a failure of the decay heat suction valve from the RB sump. Operator Action # 15 is actually an inaction step. The reference to Enclosure 11, Step 11.4 is to recognize the need to close the block valves in the discharge from EFP-2 prior to proceeding to cooldown as described in EOP-8. For Operator Action # 17, periodically evaluating the need to isolate a broken HPI line per EOP-4, "Inadequate Heat Transfer," Step 3.58, the action was not exercised in the validations performed of EOP-4 because the operator skills needing to be demonstrated for isolation of an affected HPI line were previously exercised during validations of EOP-3. See Operator Action # 5.

NRC REQUEST 5

Generally, a minimum of 80% of the operating crews should be tested (i.e., if there are six operating crews, five of the crews should be tested). Ideally, all crews should be tested on all three single failure events as a "full" crew and as a "minimum" crew. However, due to limitations in time and availability, alternative testing approaches are acceptable with adequate justification from the licensee. The objective of this testing is to assure as many crews are exposed to the required operator actions as possible, and that each single failure event is tested to demonstrate that it can be mitigated by full and minimum crew complements.

NRC REQUEST 6

Each event should be tested using a full crew and a minimum crew complement. All tests should be conducted with crews that are "naive" (i.e., have no immediate knowledge or expectation) to the single failure being tested.

FPC RESPONSE

FPC's response to both questions 5 and 6 is as follows:

Simulator training on EOP-03 and EOP-08 was conducted in accordance with simulator exercise guide ROT-9-200A (attached). The scenario set specified in this exercise guide was designed to provide each operating crew with the specific instruction, and related practice necessary to ensure that they could effectively implement the actions specified in each procedure.

As part of their training, each operating crew[1] attended comprehensive classroom training sessions covering each of the three small break LOCA solution sets and the procedure changes dealing with these solution sets (EOP-03 and EOP-08). Following the classroom presentation, each crew received an additional eight hours of simulator "training" on EOP-03 and EOP-08. During the "training" sessions the crews responded to the exercise scenarios as "unannounced casualties." These scenarios specifically address the three failure scenarios (LOBA, LOBA, EFP-2 failure) associated with TSCRN 210. Individual and crew performance were monitored by qualified instructors who intervened to provide additional training as necessary.

Evaluation exercise ROT-9-200 (attached) was developed to evaluate the effectiveness of the training provided on EOP-03 and EOP-08. In the "evaluation" mode, the scenario was implemented as an "unannounced casualty." Qualified instructors monitored and recorded details concerning individual and crew response, but did not intervene during the exercise. At the conclusion of the exercise, the crew participated in a comprehensive performance critique. All crews performed satisfactorily during the evaluation exercises.

---

[1] CR-3 has six operating crews and two backup crews

The scenario selected for the evaluation exercise involved a different, but equally complex, sequence of events from those covered during the training sessions. This approach provided assurance that operating crews could implement the requirements of EOP-03 and EOP-08 under conditions different from those specifically covered during the previous simulator training sessions.

The above training demonstrates the capability of all operating crews to handle the three scenarios covered by TSCRN-210. The ROT-9-200 Instructional Outline addresses the LOBA scenario, and the ROT-9-200-A Instructional Outline, Section 7B addresses the EFP-2 solution set and Section 7D addresses the LOBB scenario. Evidence of the above training is available in the closure documentation associated with FPC Restart Issue O-3.

GENERAL COMMENTS

The EOPs addressed by TSCRN 210 have been reviewed by the CR-3 Plant Review Committee. There have been no substantive changes made to the emergency operating procedures (EOPs) previously provided in our letters dated November 19 and 21, 1997 related to small break LOCA mitigation as a result of the review and approval process. However, there was one change made to EOP-13 Rule 2, "HPI Control," and one change made to EOP-14, Enclosure 17, "Control Complex Emergency Ventilation." The change to the proposed final version of EOP-13, Rule 2, attached, concerns one detail related to the throttling of HPI to prevent exceeding the limit of 950 psig reactor coolant system (RCS) pressure if a steam generator is isolated for a tube rupture. This setpoint was changed from 1000 psig to account for instrument uncertainty. The change to EOP-14, Enclosure 17 involved the removal of a step to bypass and reset the ES 480 V lockouts prior to starting the control complex fans. A modification is being performed which will eliminate the need for this step.

Table 1

Modifications Involving Automatic Equipment

In FPC letter dated September 25, 1997, (3F0997-30), Attachment C, Table 2, FPC identified the modifications associated with TSCRN 210. Those modifications involving automatic equipment functions and how the risk modeling addressed these modifications is discussed below.

MAR 96-11-01-1 (MOD #1)

MAR 96-11-01-1 (MOD #1) restores the automatic opening of ASV-204, the steam admission valve to EFP-2, on an "A" EFIC actuation. This modification will restore the load sharing capability of the Emergency Feedwater System for the LOCA concurrent with a LOOP and a loss of EDG-1B in order to reduce the load on EDG-1A.

The failure of this valve is already addressed in the CR-3 baseline PSA and the modification does not affect the core damage frequency.

MAR 96-12-17-01 (MOD #7)

MAR 96-12-17-01 (MOD #7) will remove the auto-start function from both nonsafety control circuits of the Flush Water Pumps. This will prevent them from auto-loading onto the EDGs.

There is no need to model this modification since these loads are not used in the mitigation of a SBLOCA.

MAR 96-06-02-01 (MOD #10)

MAR 96-06-01 (MOD #10) installs windup reset on integral controller on the EFIC system. This will provide for faster response of EFW for control of flow to the OTSGs. This reduces EFW flow and consequential EDG-1A loading upon initiation.

This modification has no impact on the EFW model already included in the CR-3 PSA since the faster response of EFW flow control does not affect the failure of EFW.

MAR 97-02-17-01 (MOD #12)

MAR 97-02-17-01 (MOD #12) changes the Engineered Safeguards automatic actuation logic for the normal Makeup supply valve MUV-27 to add automatic closure upon receipt of a diverse containment isolation signal (which also initiates HPI). The purpose of the modification is to aid in HPI flow balancing actions in the event of a broken HPI line. MUV-27 must be closed to help ensure accurate HPI flow indication.

This modification has no impact on the CR-3 PSA since the frequency of a SBLOCA/LOOP with the requirement that the SBLOCA occur in an HPI line is already very small, approximately $7x10^{-7}$ per year.

Table 2

Risk Modeling of New Operator Actions

OA	Operator Action	Failure Scenario	Basis	PSA Model	Human Error Probab-ability
9	If 'B' DC power is lost, crosstie EFP-2 to 'A' train (EFV-12) AND Secure EFP-1	LOBB	EFP-1 can only provide flow for a specific time period, then EFP-2 must be aligned.	Added 'B' side power dependency to EFP-1. Took no credit for cross-tying EFP-2 through EFV-12. EFW assumed failed given loss of 'B' DC power.	1.0
10	Put EFIC in manual permissive AND Close EFW block valves	LOBB	Required to prevent cycling of the limited duty motors on the EFW block valves. This action may be included in the EOPs for both trains of EFW.	The additional modeling associated with OA #9 assumes that EFW will fail on LOBB.	1.0
11	Manage EDG load in order to extend EFP-1 operation by - • Shutdown SWP-1A & RWP-2A after verifying redundant pumps are operating and placing switches in Pull-to-Lock to prevent reactuation of pumps (EDG loading) • Place EFP-1 Trip Defeat Switch in defeat position to prevent automatic trip of EFP-1 on RCS pressure of 500 psig	EFP-2	**Defense in Depth action** for postulated single failure of the loss of EFP-2. These actions extend the time EFP-1 is available for OTSG cooling.	Dependencies on EFP-2 operation added to SWP-1A and RWP-2A success. Operator action of placing the EFP-1 Trip Defeat Switch in defeat position given EFP-2 unavailability added to EFP-1 failure model.	1.0
14	If only EFP-2 is supplying feedwater to the OTSG, the RCS cooldown will be stopped prior to reaching an EFP-2 operational limit. Manage operation of EFP-2 by closing ASV-5 and ASV-204 on low OTSG pressure (Cycle EFW) and restart EFP-2 when pressure increases. (Mitigation strategy includes operation of diesel backed FWP-7 as a Defense in Depth action.)	LOBA LOBB	For a LOBA, or a LOBB (to manage EDG load), EFP-1 would be secured and EFW flow would rely on EFW-2. EFP-2 would be cycled due to operational limitations on low OTSG pressures.	Subsumed by failure of operator to delay cooldown and the assumed loss of EFW on loss of 'B' DC power.	1.0
15	If EFP-2 is not operating when in a LOOP condition with inadequate subcooling, limit cooldown prior to the EFP-1/LPI Interlock	EFP-2	If EFP-2 is not available, steps must be taken to ensure EFP-1 operates as long as needed.	Operator failure to delay cooldown event added to top gate of EFW model combined with failure of either of the EFPs or loss of DC power from either bus.	1.0

Table 2

Risk Modeling of New Operator Actions

OA	Operator Action	Failure Scenario	Basis	PSA Model	Human Error Probab-ability
17	Periodically re-evaluate HPI line break criteria on RCS repressurization.	LOBA LOBB EFP-2	Required for specific HPI line pinch areas to ensure a broken line will be isolated if warranted.	Not evaluated due to low frequency of SBLOCA/LOOP event with the requirement that the SBLOCA occur in an HPI line (7E-07 per year).	N/A

Table 3

Simulator Validations

OA #	ACTION	EOP / STEP	REQUIRED TIME	VALIDATED
1	TRIP RCPS < 2 MIN.	EOP-3 STEP 2.1 EOP-13, Rule 1	< 2 MINUTES	88;69;75;78;(7/30/97);(8/5/97)
2	MANUAL HPI/RBIC	EOP-13 RULE 1	< 10 MINUTES	88;69;75;78;(7/30/97);(8/5/97)
3	ENSURE 4 HPI VALVES OPEN	EOP-3 STEP 3.3	< 10 MINUTES	88;69;75;78;(7/30/97);(8/5/97)
4	ISOLATE RCP SEAL INJECTION	EOP-3 STEP 3.8	< 20 MINUTES	88;69;75;78;(7/30/97);(8/5/97)
5	ISOLATE BROKEN HPI LINE	EOP-3 STEP 3.6 EOP-3, Step 3.10	< 20 MINUTES	88;69;75;78; (7/30/97);(8/5/97)
6	ENSURE EFIC ACTUATES	EOP-3 STEP 3.9	< 20 MINUTES	88;69;75;78; (7/30/97);(8/5/97)
7	START CONTROL COMPLEX VENTILATION	EOP-3 STEP 3.12 DIRECTS USE OF EOP-14, ENCLOSURE 17	< 30 MINUTES	88;69;75;78; (7/30/97);(8/5/97)
8	TRANSFER BWST TO RB SUMP	EOP-3 STEP 3.13 DIRECTS USE OF EOP-14, ENCLOSURE 19	>20 MINUTES	Not validated on simulator for small breaks. See answer to Request 4.
9	CROSS TIE EFP-2 TO A-TRAIN AND SECURE EFP-1	EOP-3 STEP 3.16 DIRECTS USE OF EOP-14, ENCLOSURE 11 (STEP 11.7)	> 20 MINUTES	88;78; (7/30/97);(8/5/97)
10	PLACE EFIC IN MANUAL PERMISSIVE AND CLOSE EFW BLOCK VALVES	EOP-3 STEP 3.16 DIRECTS USE OF EOP-14, ENCLOSURE 11 (STEP 11.4)	> 20 MINUTES	88;78;(7/30/97)(8/5/97)
11	MANAGE EDG LOADS: S/D SWP-1A & RWP-2A EFP-1 TRIP DEFEAT	EOP-3 STEP 3.16 DIRECTS USE OF EOP-14, ENCLOSURE 11 (STEP 11.12;11.13;11.14)	> 20 MINUTES	88;78:(7/3097);(8/5/97)

Table 3

Simulator Validations

OA #	ACTION	EOP / STEP	REQUIRED TIME	VALIDATED
12	START CONTROL COMPLEX CHILLER	EOP-3 STEP 3.17 and EOP-8 STEP 3.8 DIRECTS USE OF EOP-14, ENCLOSURE 18	< 80 MINUTES	111; 69 (running); 75;76;78;88;(7/30/97);(8/5/97)
13	STOP RB SUMP PUMPS	EOP-8 STEP 3.11;3.12	> 20 MINUTES	111;78;69;75;76
14	IF ONLY EFP-2 IS AVAILABLE THEN STOP COOLDOWN BEFORE REACHING OPERATIONAL LIMIT OF EFP-2	EOP-8 STEP 3.17 DIRECTS USE OF EOP-14, ENCLOSURE 7 (STEP 7.16)	> 20 MINUTES	78;69
15	IF EFP-2 IS NOT OPERATING, LIMIT COOLDOWN PRIOR TO EFP-1/LPI INTERLOCK	EOP-3 STEP 3.16, IN CONJUNCTION WITH EOP-14, ENCLOSURE 11, IS A PERFORM STEP WHICH MUST BE COMPLETED PRIOR TO PROCEEDING TO COOLDOWN GUIDANCE IN EOP-8. (See Step 11.14)	> 20 MINUTES	(8/5/97)
16	ESTABLISH COOLDOWN USING TBVs AND ADVs	EOP-8 STEP 3.19	> 20 MINUTES	78;69
17	PERIODICALLY RE-EVALUATE HPI LINE BREAK ISOLATION CRITERIA ON RCS REPRESSURIZATION	EOP-4 STEP 3.58	> 20 MINUTES	See Operator Action # 5

Unnumbered -SBLOCA with LOBB (7/3/97)
Unnumbered -SBLOCA with LOBB (8/5/97)
69 - Cooldown with LOBA (8/5/97)
111 - SBLOCA & EFP-2 Failure (11/21/97)
75 - Loss of SCM with No EFW and degraded HPI (8/13/97)
76 - HPI/PORV Cooling to LOCA Cooldown (EFP-2 Failure) (8/14/97)
78 - LOBB with Cold Leg SBLOCA (8/20/97)
88 - SBLOCA/LOCA/LOBB (9/15/97)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

ECP-13, Rules

EOP RULES

Rule 1, Loss of SCM

Rule 2, HPI Control

Rule 3, EFW Control

Rule 4, PTS

Rule 5, EDG Control

Approved by MNPO ____________________ Date __________
(SIGNATURE ON FILE)

RULE 1, LOSS OF SCM

Adequate SCM

RCS	SCM
> 1500 psig	≥ 30°F
≤ 1500 to > 350 psig	≥ 50°F
≤ 350 psig	SPDS
≤ 160°F	N/A

- ___ IF < 2 minutes have elapsed since losing adequate SCM, THEN trip all RCPs.

- ___ IF RCPs were NOT tripped within 2 minutes, THEN ensure 1 RCP remains running in each loop until SCM is restored or LPI flow is > 1400 gpm in each injection line.

- ___ IF a running RCP trips, THEN bypass start permissives using key 50 and start the other RCP in that loop.

- ___ Depress "HPI MAN ACT" push button on Train A and B.

- ___ Depress "RB ISO MAN ACTUATION" push button on Train A and B.

- ___ IF LPI has NOT actuated, AND RCS PRESS ≤ 300 psig, THEN depress "LPI MAN ACT" push button on Train A and B.

RULE 2, HPI CONTROL

• ___ IF HPI has actuated, THEN bypass or reset ES actuation.	1 ___ Obtain SRO concurrence to bypass or reset ES. 2 Bypass or reset ES actuation: ___ Auto ___ Manual
• ___ IF adequate SCM exists, THEN throttle HPI to prevent exceeding limits.	• ___ NDT limit • ___ PTS • ___ RCS PRESS ≤ 950 psig (if OTSG is isolated for tube rupture)
• ___ HPI may be throttled any time adequate SCM exists based on $T_{incores}$.	
• ___ Open MUP recirc prior to throttling HPI flow < 200 gpm/pump.	• ___ IF aligning to MUT, THEN open MUP recirc to MUT valves: ___ MUV-53 ___ MUV-257 • ___ IF aligning to RB sump, THEN open HPI recirc to sump valves: ___ MUV-543 ___ MUV-544 ___ MUV-545 ___ MUV-546

RULE 3, EFW CONTROL

Required OTSG levels

"LLL"	> 20 in	≥ 1 RCP running with adequate SCM
"NAT CIRC"	> 70%	No RCPs running with adequate SCM
"ISCM"	> 90%	Inadequate SCM

NOTE

EFW is not required if LPI flow > 1400 gpm in any line.

- Inadequate SCM exists.

___ IF OTSG levels are NOT progressing towards the "ISCM" setpoint, THEN take manual control.

- EFW flow required for manual control:

2 OTSGs	> 280 gpm in 1 line to each OTSG
1 OTSG	> 470 gpm in 1 line to 1 OTSG

- Adequate SCM exists.

___ Throttle EFW to prevent OTSG PRESS from lowering > 100 psig below desired PRESS.

- Do not allow OTSG level to lower.

- ___ IF any EFW control valve fails to operate, THEN control EFW flow.

1 ___ Depress "MANUAL PERMISSIVE" push buttons on EFIC channels A and B.

2 ___ Close EFW block valve to isolate any failed control valve.

3 ___ De-energize any EFW block valve that was closed.

4 ___ Actuate EFIC.

RULE 4, PTS

- PTS is in effect if <u>any</u> of the following conditions exist:

  ___ T_{cold} < 380°F and cooldown rate exceeds ITS limit

  ___ RCPs off and HPI flow exists

- ___ <u>IF</u> PTS is in effect, <u>THEN</u> perform required actions.

- ___ Throttle HPI flow to minimize adequate SCM.

- ___ Throttle LPI flow to minimize adequate SCM.

- ___ <u>IF</u> cooldown is required, <u>AND</u> cooldown rate can be controlled, <u>THEN</u> maintain cooldown rate within ITS limit.

- ___ PTS is applicable until an Engineering evaluation has been completed.

RULE 5, EDG CONTROL

Maximum EDG Load Limits

Starting Load	Running Load
3884 KW	3374 KW

- IF manually applying load to the EDG, THEN ensure existing EDG load is < Max Allowable Load prior to starting component:

Component	Max Allowable Load (BSP shutdown)	Max Allowable Load (BSP running)
AHF-1A/B/C	2927	3313
AHF-17A/B	2969	3324
AHF-18A/B	2969	3324
AHF-19A/B	3034	3356
AHF-54A/B	3050	3361
BSP-1A/B	3078	N/A
CHP-1A/B	3040	3355
CHHE-1A/B	2665	3178
DCP-1A/B	2883	3295
DHP-1A/B	2340	3090
EFP-1	1843	2649
MUP-1A/B/C	1644	2450
RWP-2A/B	1817	2623
RWP-3A/B	23[illegible]8	3134
SWP-1A/B	1891	2697
SFP-1A/B	2987	3332

3.6.5 Fast Start: Verify the diesel starts from standby conditions and achieves, in ≤ 9.5 seconds, voltage and frequency as follows:

IF the administrative range is exceeded,
THEN entry into LCO 3.8.1 is required.

EDG Voltage Range Limits	Minimum	Maximum
Technical Specification	3933 Volts	4400 Volts
Administrative (Accuracy Corrected)	4100 Volts (117.2 Volts measured downstream of PT)	4220 Volts (120.5 Volts measured downstream of PT)

IF the administrative range is exceeded,
THEN entry into LCO 3.8.1 is required.

EDG Frequency Range Limits	Minimum	Maximum
Technical Specification	58.8 Hz	61.2 Hz
Administrative	59.4 Hz	60.6 Hz

3.6.6 Verify the EDG operates for ≥ 60 minutes at the following KW load ranges (must be preceded by a successful start):

EDG Loading Range Limits	Minimum	Maximum
Technical Specification	2,600 KW	2,850 KW
Administrative (Accuracy Corrected)	2,625 KW	2,825 KW

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT C

EOP-14, Enclosure 17, Control Complex Emergency Ventilation

ENCLOSURE 17 CONTROL COMPLEX EMERGENCY VENTILATION

ACTIONS	DETAILS
17.1 ___ Align control complex ventilation in recirc.	• Select "CONTROL COMPLEX HVAC ISOLATE RESET" switches to "ISO": ___ A Train ___ B Train
17.2 ___ Verify control complex is isolated.	• Verify the following dampers are closed:

___ AHD-1C
___ AHD-1E
___ AHD-2C
___ AHD-2E
___ AHD-12
___ AHD-12D

17.3 ___ Ensure ventilation fans are shutdown.

A Train	B Train
___ AHF-17A	___ AHF-17B
___ AHF-19A	___ AHF-19B

ACTIONS	DETAILS

NOTE

Performance of the next step ensures adequate EDG load margin available to start all fans required by this enclosure.

17.4 ___ IF EDG Bkrs are closed, THEN verify EDG load is < max allowable load.

Max Allowable Load (BSP running)	3280 KW
Max Allowable Load (BSP shutdown)	2960 KW
A EDG Load	____ KW
B EDG Load	____ KW

NOTE

B Train fans are preferred for EDG load concerns.

17.5 ___ Establish CC ventilation in emergency recirc.

1 ___ Ensure AHD-3 is open.

2 Ensure only one train of CC ventilation running:

A Train	B Train
___ AHF-18A	___ AHF-18B
___ AHF-19A	___ AHF-19B

ACTIONS	DETAILS
17.6 ___ Ensure 1 EFIC fan running.	• ___ IF starting AHF-54A, THEN perform the following: ___ Select "TEMP CONT. VV, CHV-113" switch to "MOD" position. ___ Start AHF-54A • ___ IF starting AHF-54B, THEN perform the following: ___ Select "TEMP CONT. VV, CHV-100" switch to "MOD" position. ___ Start AHF-54B
17.7 ___ Establish ventilation for chemistry sampling.	• Ensure only one train of ventilation running:

A Train	B Train
___ AHF-20A in "SLOW"	___ AHF-20B in "SLOW"
___ AHF-44A	___ AHF-44B

ACTIONS	DETAILS
17.8 ___ Notify PPO to ensure chill water is aligned to running fan.	• ___ IF AHF-18A is running, THEN ensure the following alignment: ___ CHV-2 "CC Cooler B Outlet Iso" is closed (164 ft CC by Ventilation Room door) ___ CHV-4 "CC Cooler A Outlet Iso" is open (164 ft CC between AHHE-5A and AHHE-5B) • ___ IF AHF-18B is running, THEN ensure the following alignment: ___ CHV-4 "CC Cooler A Outlet Iso" is closed (164 ft CC between AHHE-5A and AHHE-5B) ___ CHV-2 "CC Cooler B Outlet Iso" is open (164 ft CC by Ventilation Room door)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT D

Instructional Outlines, ROT-9-200 & ROT-9-200A