Florida Power
CORPORATION

December 17, 1982
3F-1282-19

Mr. John F. Stoltz, Chief
Operating Reactors Branch #4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
NUREG-0737, Item II.E.1.1
Auxiliary (Emergency) Feedwater System Evaluation

Dear Mr. Stolz:

On November 19, 1982, Florida Power Corporation (FPC) responded with the additional information on the subject evaluation which you requested on August 19, 1982, except Items 4, 8, and 10. The response to Item 10 is attached.

Items 4 and 8 requested FPC to commit to modifications of the Crystal River Unit 3 Emergency Feedwater System which are beyond the scope of the Emergency Feedwater Initiation and Control (EFIC) Upgrade which will fulfill the requirement of NUREG-0737, Item II.E.1.2, Emergency Feedwater Upgrade. For that reason, the evaluation of these modifications and their interaction with the new EFIC Upgrade is requiring longer than expected reviews. The response to Item 4 and 8 will be submitted upon resolution within FPC.

Very truly yours,

G. R. Westafer
Manager
Nuclear Licensing and Fuel Management

Attachment

Bright(F04)C3-2

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NRC QUESTIONS ON THE EFW SYSTEM AT CR-3

Reference: NRC Letter to FPC, Reid to Hancock, NUREG-0737 Item II.E.1.2, Emergency Feedwater System Upgrade, January 28, 1981.

Question 1.a.

Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1. Loss of main feed (LMFW)
2. LMFW w/loss of offsite AC power
3. LMFW w/loss of onsite and offsite AC power
4. Plant cooldown
5. Turbine trip with and without bypass
6. Main steam isolation valve closure
7. Main feedline break
8. Main steam line break
9. Small break LOCA
10. Other transient or accident conditions not listed above.

Response to Question 1.a.

The original design of the Emergency Feedwater (EFW) System established a requirement for a minimum flow of 740 gpm as reported in Chapter 10 of the FSAR. Where appropriate, this flow rate is used as a part of the transient analysis for the accidents considered in the FSAR. Table 1 contains a list of those transients considered in the FSAR along with the acceptance criteria.

As a part of the EFW upgrade, the requirements for minimum flow were re-examined. The design basis event for the EFW system is LMFW with a concurrent loss of offsite power (LOOP), and subsequently loss of reactor coolant pumps. The pertinent parameters for this accident relative to the EFW system are design flow rate and required time to full EFW flow. These parameters reflect the functional requirements of the EFW system to a) remove decay heat, and b) provide a smooth reactor coolant flow transition from reactor coolant pump operation to natural circulation. An evaluation of the loss of main feedwater transient for the EFW system has been performed in detail for the Rancho Seco plant which is similar in primary system design to Crystal River III (CR-3). By comparative analysis it has been concluded that the results of the Rancho Seco analysis are applicable, conservative, and bounding for the CR-3 plant. These analyses show that a minimum EFW flow rate of 740 gpm for CR-3 and a 50 second delay between the initiating signal and achieving minimum flow is acceptable and adequate to meet the acceptance criteria. The analysis and assumptions are described in response to Questions 1.b and 2.

Accidents 1 through 4 of Table 1 specifically require EFW for mitigation. The results of these analysis were acceptable and are described in the FSAR sections noted in Table 1. The other accidents listed in Table 1 do not require EFW for mitigation although the availability of the EFW system is assumed.

The other events listed in the question but not included in Table 1 are discussed below:

LMFW with Loss of Onsite and Offsite AC Power - This event was not a design basis for the plant and subsequently is not included in Chapter 14 of the FSAR. Although a specific analysis of the event is not included, the upgraded EFW system is designed to supply at least 740 gpm even with the loss of both onsite and offsite AC power.

Plant Cooldown - Plant cooldown with EFW is a controlled event with decay heat levels equal to or lower than the loss of feedwater event identified as the design basis event. The design basis event bounds this case for EFW flow required.

Turbine Trip with and without Bypass - This event does not affect the EFW system unless MFW fails. In which case, the LMFW event previously addressed would bound the EFW system design.

Main Steam Isolation Valve Closure - This event does not directly affect the EFW system unless MFW is lost as discussed above.

Question 1.b.

Describe the plant protection acceptance criteria and corresponding technical basis used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel control temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage.
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cooldown the primary system.

Response to Question 1.b.

The design basis event for sizing the EFW system is the loss of Feedwater Event for which the acceptance criteria are:

RCS Pressure < 110% of design
DNBR > Applicable correlation limit
Doses < 10CFR100

The acceptance criteria for the other transients which assume the availability of EFW are given in Table 1.

The acceptance criteria for these accidents include RCS pressure limits, fuel limits and offsite dose limits. The RCS cooling rate is not an acceptance criterion for accident analyses. An overcooling event that drains the pressurizer is not desirable, however, it does not violate any of the accident analysis acceptance criteria.

Maintaining a minimum steam generator level is also not an acceptance criterion for accident analyses. It is desirable that the reactor is tripped and EFW initaited prior to steam generator dryout, but this is not required in order to obtain acceptable results. After EFW has been initiated, the high injection point in the steam generator reduces system dependence on a specific level for adequate heat transfer. The steam generator level control is set low for decay heat removal with forced circulation and high for natural circulation. The level is even higher for small break LOCA events.

TABLE 1

	Accident Description	FSAR Section	Acceptance Criteria
1.	Loss of Electric Power	14.1.2.8	A, B, D
2.	Main Steamline Failure	14.2.2.1	D
3.	Main Feedline Break	14.2.2.9	A
4.	Loss of Coolant Flow	14.1.2.6	A, B
5.	Startup Accident	14.1.2.2	A, B
6.	Rod Withdrawal Accident at Rated Power Operation	14.1.2.3	A, B
7.	Moderator Dilution Accident	14.1.2.4	A, B
8.	Cold Water Accident	14.1.2.5	A, B
9.	Stuck Out, Stuck In, or Dropped Control Rod Accident	14.1.2.7	A, B
10.	Steam Generator Tube Failure	14.2.2.2	D
11.	Rod Ejection Accident	14.2.2.4	C, D
12.	Loss of Coolant Accident and Small Break LOCA	14.2.2.5	D, E

Key	Acceptance Criterion	Technical Basis
A	RCS Pressure < 110% of Design	ASME Code
B	DNBR > Applicable Correlation Limit	SRP 4.4
C	280 Cal./Gram Fuel Limit	RG 1.77
D	Doses < 10CFR100	10CFR100
E	Fuel Cladding < 2200°F	10CFR50.46

Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above.

Response to Question 2

As discussed in response to Question 1.a., the design basis event which verifies the EFWS design flowrate is loss of main feedwater. The analysis assumptions for this event are listed below. Corresponding technical justification, where not specifically listed, is based on licensing requirements and prudent engineering judgement at the time of the analysis. The informations is not provided for the other events identified in Question 1.a. and Table 1 because the LMFW event is the most limiting.

a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   - 102% full power; this includes an allowance for 2% power level measurement uncertainty.

b. Time delay from the initiating event to reactor trip.

   - The reactor will trip on high RC pressure approximately 14 to 15 seconds after the LMFW event.

c. Plant parameter(s) which initiates EFWS flow and time delay between initiating event and introduction of EFWS flow into steam generator.

   - The EFWS will be initiated by the low OTSG level trip. The time delay between receiving the initiate signal and full EFW flow to the steam generators is 50 seconds for a case with the loss of offsite power. This is a total delay of approximately 64 seconds from the LMFW event.

d. Minimum steam generator water level when initiating event occurs.

   - Steam generator inventory rather than water level is used as an input to this analysis. (See 2e)

e. Initial steam generator water inventory and depletion rate before and after EFWS commences - identify reactor decay heat rate used.

   - The intial steam generator inventory is dependent on power level. For this case, a liquid inventory of 39,600 lbm per steam generator was used. The depletion rate of the inventory decreases after initiation of EFW and depletion continues until the liquid inventory is essentially depleted at about 80 seconds. From this point, the entire EFW flow is vaporized until decay heat plus RC pump heat drops below the capability of the EFW system. At that time, steam generator inventory begins increasing again. The decay heat used in this calculation was 1.2 times ANS 5.1 decay heat.

f. Maximum pressure at which steam is released from the steam generator(s) and against which the EFW pump must develop sufficient head.

   - The peak steam pressure occurs shortly after the initiating event and is less than 1100 psig. Soon after the EFW is initiated, however, the steam pressure is controlled by the first bank of steam safety valves to a pressure of about 1050 psig.

g. Minimum number of steam generators that must receive EFW flow.

   - This analysis was run assuming both steam generators were available. The heat load can be removed with one OTSG and it is expected that the results would be approximately the same.

h. RC flow condition - continued use of RC pumps or natural circulation.

   - Continued operation of RC pumps was used for this analysis.

i. Maximum EFW inlet temperature.

   - An inlet temperature of 120°F was used.

j. Following a postulated steam or feedline break, time delay assumed to isolate break and direct EFW flow to intact steam generator(s). EFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also, identify credit for primary system heat removal due to blowdown.

   - FSAR Sections 14.2.2.1 and 14.2.2.9 contain the details of the assumptions used in the main steam line and feedline break analyses.

k. Volume and maximum temperature of water in main feedlines between steam generator(s) and EFWS connection to main feedline.

   - There are no piping connections between the EFWS and the MFW system.

l. Operating condition of steam generator normal blowdown following initiating event.

   - The OTSG's do not have blowdown system.

m. Primary and secondary system water and metal sensible heat used for cooldown and EFW flow sizing.

   - Plant cooldown was not considered in the design basis analysis. A primary system sensible heat of 1.256×10^{6} BTU/^{o}F was used for calculating the volume of feedwater required to cool the RCS to decay heat system parameters.

n. Time at hot standby and time to cool down the RCS to DHR system cut-in temperature to size EFW water source inventory.

- The condensate storage tank is sized to accommodate the plant at hot shutdown for about eight hours followed by a cooldown to the decay heat system cut-in temperature of 280°F.

Question 3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Response to Question 3

The original design criteria from Babcock & Wilcox (B & W) required each Emergency Feedwater Pump to be sized to furnish a flow capacity of 740 gallons per minute (gpm), whereby 720 gpm is delivered to both steam generators for decay heat removal and 20 gpm for pump recirculation. On February 27, 1981, B & W advised FPC that a minimum flow rate of 500 gpm is acceptable to maintain the decay heat removal from the steam generators (CR-81-028, R. J. Finnin to D. C. Poole).

Based on this, each Emergency Feedwater Pump has been adequately sized to permit a minimum flow of 500 gpm with 240 gpm for recirculation flow, seal leakage and pump wear. At the present time, the Emergency Feedwater System has no flow restriction orifices installed in either of the recirculation lines from the pumps to the Condensate Storage Tank. Actual tests have shown a flow in each recirculation line of 190 gpm. If flow restriction orifices are ever installed in the pump recirculation lines, the flow to the steam generators for decay heat removal will be increased.