Florida Power
CORPORATION

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Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit No. 3
Document No. 502-302
Operating License No. DPR-72
3-Pump Operation With Reactor Coolant Pump Power Monitors Bypassed

Dear Mr. Denton:

Subsequent to numerous requests to operate Crystal River Unit 3 at various power levels with and without Reactor Coolant Pump Power Monitors (RCPPM's), Florida Power Corporation (FPC) hereby supplements our letters to you dated April 9, 1982, and May 28, 1982. We will provide the needed information requested in your July 16, 1982 letter to us. Our intent in this letter is to clarify the outstanding issue of 3-pump operation for Crystal River Unit 3 (CR-3) by providing a comprehensive discussion of historical background, use of flux/flow/imbalance trip functions, and error assumptions used in thermal-hydraulic analyses. Your expeditious review and approval of 3-pump operation at CR-3 is needed to enhance operational flexibility and assure continued unit availability.

FPC began pursuing operation of Crystal River Unit 3 at the FSAR design power level (2544 MWt) shortly after the unit began commercial operation. Final NRC approval following an April 1981 ACRS review was completed in June 1981 (Reference 1). FPC elected to delay implementation until Refuel III (Fall 1981) due to installation difficulties associated with the RCPPM circuitry (Reference 2). The RCPPM's were mandated by the NRC to reduce the time delay before reactor tip on loss of flow based on flux/flow/imbalance thereby assuring adequate Minimum Departure from Nuclear Boiling Ratio (MDNBR). This reduced the time delay from approximately 1.4 seconds to 0.62 seconds. This produced a Cycle IV MDNBR well in excess of 2.0 (vs. a 1.3 acceptance criteria).

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Following restart from Refuel III, the unit experienced a series of spurious reactor trips, during transformer swap-overs, reactor coolant pump starts, etc. It was determined that many anticipated perturbations on the 6.9KV busses, which power the reactor coolant pumps (RCP's), caused current/voltage transients of sufficient duration to cause a reactor trip. These spurious trips caused two adverse effects: overall unit output dropped rather than increased as a result of power level upgrade and plant safety equipment was subjected to repeated, unnecessary challenges.

In April 1982, FPC management decided to dedicate extensive efforts to eliminate the RCPPM trip. A series of power escalations were proposed (References 4, 5, and 6) and approved. The first level approved (75% Full Power) was based on engineering judgement (Reference 7). The second (approximately 90% Full Power) was based on an analysis retaining all Cycle IV parameters to within original design requirements (Reference 8). The third (approximately 97%) was based on maximizing Cycle IV output via a complete core analysis without credit for the RCPPM's (Reference 9). All of the approvals to date were for 4-pump operation only.

FPC became increasingly concerned about maintaining three (3) RCP operational flexibility when a breaker problem on an RCP forced the unit off line (Reference 10). However, insufficient information had been presented by FPC to convince the NRC staff to grant 3-pump approval (Reference 9). Prior to the "final" power level request, FPC had understood with B&W concurrence (Reference 10) that the linear reduction in flux/flow/imbalance trip setpoint for 3-pump operation from the 4-pump would provide adequate DNBR protection. This action was believed to be conservative. Immediately, prior to the request to operate at 2475 MWt, B&W notified FPC of an additional complication caused by the Reactor Protection System (RPS) "instrument inaccuracy" concerns raised by B&W in their review of Environmental Qualification data (Reference 11). The 3-pump analysis was shown to be more dependent on errors (due to calculational dependencies on pressures) than the 4-pump coastdown and the time delay before trip, is longer for 3-pump coastdowns. At this point, a better understanding of how the flux/flow/imbalance trip works may be helpful.

The "barn-roof" shape of this trip envelope as represented in the Technical Specifications curves (See Figures 2.1.2 and 2.2.1) is somewhat misleading. The RPS utilizes only three sets of parameters for this

trip: flux-to-flow ratio; slopes (positive and negative), and breakpoint (positive and negative). The two curves shown in Technical Specifications simply represent two of an infinite array of trip setpoints based on varying flows. On a loss of flow transient, the "trip line" falls with decreasing flow until it "catches" reactor power and causes a reactor trip. This time delay is an important parameter in the thermal-hydraulic analysis. This time delay and the initial power become the critical parameters in the analysis of the reduction in cooling, caused by the reducing flow.

The original Cycle 4 analysis employed the pump power monitor trip. The pump power monitor senses whether three (3) or more RCP's are operating; if not, a reactor trip is caused. The time delay of a pump power monitor trip is the same for a 4 or a 3-pump coastdown, because the trip is initiated upon the loss of the pumps. With a constant time to trip, the 3-pump coastdown initial power level can be maintained at 75% of full power without violation of MDNBR criteria.

Cycle 4 (without RCPPM's) was subsequently analyzed to determine the maximum power level at which the flux/flow/imbalance trip function can be used to achieve an acceptable DNBR response during a loss of flow transient for both 4-pump and 3-pump operation. This analysis is the basis for Technical Specification Change Request No. 98 (Reference 6).

The 3-pump coastdown differs from the 4-pump coastdown in that a reactor trip is generated at a later time during the 3-pump coastdown. This later trip time is a combination of two effects.

1) A constant flux/flow ratio is maintained for both 4 and 3-pump operation.

2) The reactor coolant flow error increases as flow decreases. A later trip time results in a lower flow at time of trip. Since reactor power is assumed to be constant until trip, the initial value of reactor power is reduced to accommodate the lower flow at time of trip. Analysis of the 3-pump coastdown has shown that an initial power level of 55% of 2544 MWt (Full Power) results in acceptable DNBR response during the 3-pump coastdown.

The pump coastdown analysis includes RPS setpoint uncertainties in both power (flux) and reactor coolant flow. The power uncertainty was accounted for in the initial conditions while the flow uncertainties were accounted for in the Technical Specification setpoint calculation. The specifics of these uncertainties are discussed below.

Power Uncertainity: A 6% uncertainty was accounted for in the initial transient conditions. The flux uncertainty was necessary as the flux/flow/imbalance trip uses a flux measurement to actuate a reactor trip. The uncertainty accounts for the following terms:

1)	Heat Balance Error	2% FP
2)	Steady-state Neutron Measurement Error	2% FP
3)	Transient Induced Neutron Measurement Error	2% FP

Flow Uncertainty: Flow uncertainties are accounted for when determining the Technical Specification flux/flow/imbalance setpoint from the accident analysis setpoint of 1.0139 %FP/% flow for these transients. A flow error is determined for each coastdown based on the minimum expected flow. This is necessary to account for an increasing flow error with decreasing flow. The RPS instrument error used for the 4 and 3-pump coastdowns are 3.78% and 4.22%, respectively.

In addition, flow noise of 1.5% flow is also accounted for in determining the Technical Specification flux/flow/imbalance setpoint. This assumption minimizes the flow at the time of trip, thereby providing a more conservative trip condition.

The initial conditions and results derived from the analysis are presented in Table 1 and Figure 1.

Florida Power Corporation is anxious to meet with your staff in clarifying and assuring timely review and approval of 3-pump operation. Please advise Mr. David G. Mardis, Acting Manager, Nuclear Licensing, at (813) 866-4283 of an appropriate time for a meeting.

Very truly yours,

John A. Hancock
Vice President
Nuclear Operations

Mardis(F01)ND26-3

cc: Mr. J. P. O'Reilly
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
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TABLE 1
CRYSTAL RIVER 3
DATA FOR LOSS OF FLOW TRANSIENT ANALYSIS

4 Pump Coastdown curve (Flow vs time) - Figure 1
3 Pump Coastdown curve (Flow vs time) - Figure 2
DNBR vs time 4 Pump/3 Pump* - See note below

Sequences of Events:

Loss of Power to RCP	4 Pump @ time = 0.0	3 Pump @ time = 0.0
Flux/Flow setpoint, RPS Setpoint	1.0139	1.0139
Calculated Limit	1.071	1.126
Rod Motion Begins	3.2 seconds	5.85 seconds
Minimum DNBR Number	1.35**, @ 4.2 sec.	1.35**, @ 6.6 sec.

	Initial Value	Uncertainty	Total Value
Power, % 2544 MWt (Includes 2% FP for control band)***			
4 pump	99.3%	6% FP	105.3% FP
3 pump	57.0	6% FP	63.0% FP
Initial Flow, 4 pump, Minimum design flow			106.5% of 88000 GPM/pump
3 pump		74.7% of	" " " " "
Tin, °F			
4 pump	554.7 F	+2 F	556.7
3 pump	559.3	+2 F	561.3
Pressure	2200 psia	-65 psia	2135 psia
Radial Peaking			
4 pump			1.671
3 pump			1.714
Axial Shape			1.5 cosine
Initial Steady-State DNBR, 4 pump			2.45
3 pump			3.9

RPS Setpoint Uncertainties: See attached discussion

* DNBR vs time curves were generated for several power levels in an iteration process to determine the maximum power level that satisfied the minimum DNBR criteria of 1.35. The final power level was determined by interpolation so an exact DNBR vs time curve was not generated.

** Based on a CHF correlation design limit of 1.30 plus .05 to account for the rod bow DNBR penalty applied to Cycle 4 operation.

*** Initial power value used in determining setpoint values. Total power value used in core thermal-hydraulic analyses.

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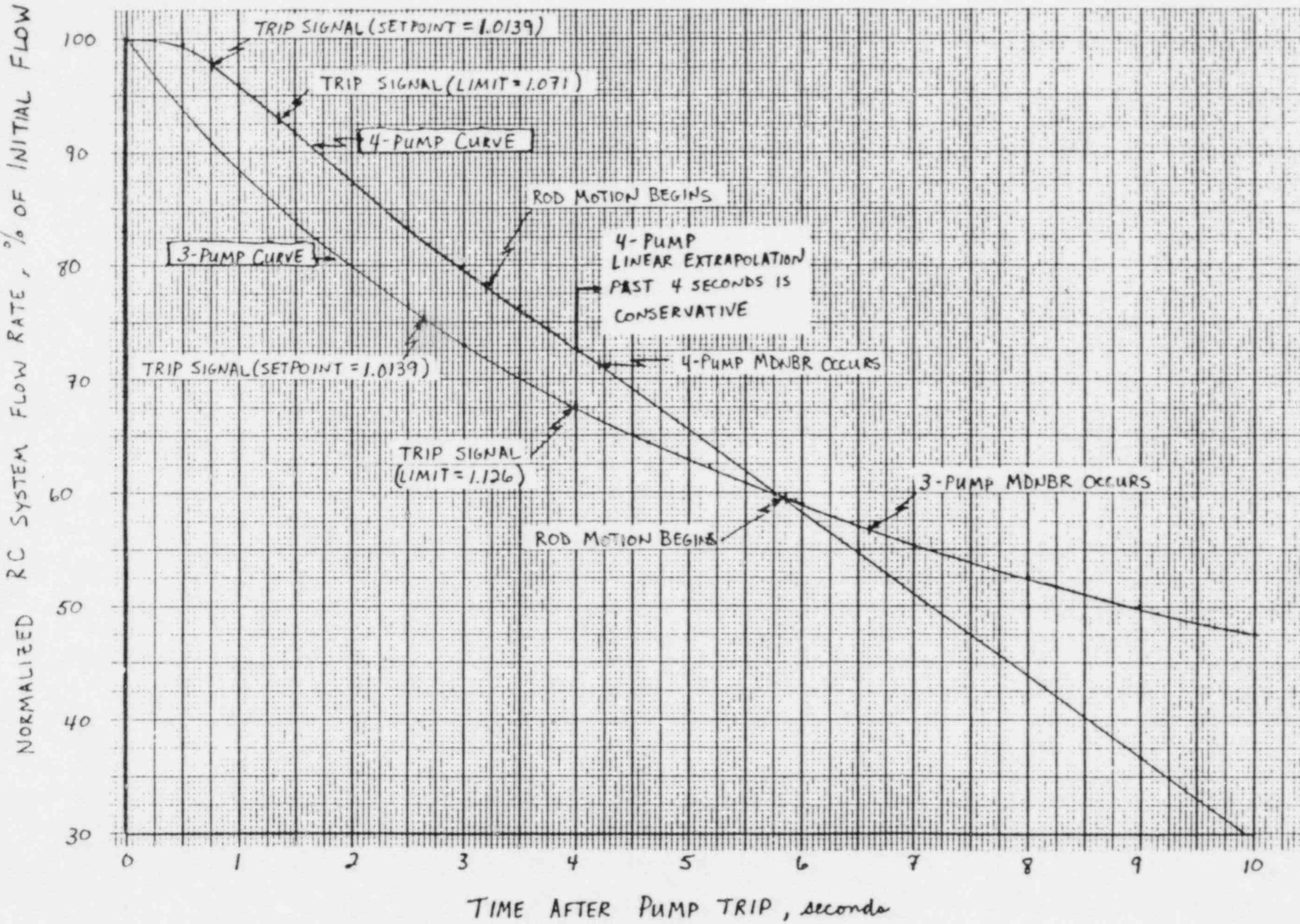
4-PUMP AND 3-PUMP COASTDOWN CURVES FOR CRYSTAL RIVER 3



REFERENCES

1. Amendment 41 to Facility Operating License DPR-72; Letter from John F. Stolz, NRC to John A. Hancock, FPC; dated July 21, 1981.

2. Technical Specification Change Request No. 70, Supplement 1 W. A. Cross, FPC, to Darrell G. Eisenhut, NRC; dated July 27, 1981.

3. Amendment 42 to Facility Operating License No. DPR-72; Letter from John F. Stolz, NRC to John A. Hancock, FPC; dated August 12, 1981.

4. Request For Interim Emergency Authorization Letter from P. Y. Baynard, FPC to Harold R. Denton, NRC: dated April 1, 1982.

5. Request for Authorization to Operate at up to 2300 MWt; Letter from D. G. Mardis, FPC, to Harold R. Denton, NRC; dated April 6, 1982.

6. Technical Specification Change Request No. 98 Letter from D. G. Mardis, FPC, to H. R. Denton, NRC; dated May 28, 1982.

7. Confirmation of April 1, 1982 Verbal Authorization to Operate at 75% Full Power. Letter from T. M. Novak, NRC, to John A. Hancock, FPC; dated April 2, 1982.

8. Confirmation of April 6 VerBal Authorization to Operate at 2300 MWt Letter from T. M. Novak, NRC to John A. Hancock, FPC; dated April 16, 1982.

9. Amendment No. 56 to Facility Operating License DPR-72 Letter from Sydney Miner, NRC to John A. Hancock, FPC; dated July 16, 1982.

10. Three Pump Operation Without RCPPM's Letter from D. G. Mardis, FPC to Harold R. Denton, NRC; dated April 9, 1982.

11. BAW-10003, "Qualification Testing of Protection System Instrumentation".