Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

August 26, 1992

3F0892-06

U.S. Nuclear Regulatory Commission
Attn: Document Control Room
Washington, D.C. 20555

Subject: Bulletin 88-04: Potential Safety-Related Pump Loss
Decay Heat Removal Pump Mission Time

Dear Sir:

Florida Power Corporation (FPC) is providing a supplemental response to NRC Bulletin 88-04 as Attachment 2 to this letter. This response documents our recent simulator evaluation of Decay Heat Pump mission time which provides closure to this issue. FPC continues to conclude there is adequate assurance that the DH pumps would perform their intended function during all design basis accidents. This conclusion is based upon Loss of Coolant Accident analysis, which draws upon NRC approved models, results from actual pump testing, and two independent simulator evaluations.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/JWT

Attachments

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

IE37
1/1

ATTACHMENT 1

STATE OF FLORIDA

COUNTY OF CITRUS

P. M. Beard, Jr. states that he is the Senior Vice President, Nuclear Operations for Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this day of August, 1992.

Notary Public

Notary Public, State of Florida at Large,

My Commission Expires

Notary Public, State of Florida at Large
My Commission Expires Dec. 18, 1995
~~Bonded thru Agent's Notary~~ Brokerage

ATTACHMENT 2

REFERENCES

A. NRC to FPC letter 3N0492-14, dated April 29, 1992
B. FPC to NRC letter 3F0789-04, dated July 11, 1989
C. NRC to FPC letter 3N0889-24, dated August 24, 1989
D. NUREG/CR-5706, dated June 1991
E. FPC to NRC letter 3F1189-02, dated November 1, 1989

BACKGROUND

Bulletin 88-04, "Potential Safety-Related Pump Loss", was issued on May 5, 1988. The Bulletin required utilities to evaluate the adequacy of the minimum flow bypass lines with respect to damage resulting from operation and testing in the minimum flow mode. The original minimum bypass flow design requirement for the CR-3 DH pumps was 80-100 GPM. As part of the Bulletin 88-04 investigative effort, FPC contacted the DH pump vendor, regarding the question of minimum bypass flow. The CR-3 pumps were originally designed and manufactured by Worthington Corporation. However, Worthington sold the DH pump vendor responsibility to Dresser Corporation several years ago.

Dresser's response to FPC was that the minimum pump flow should be no less than 1200 GPM for up to 100 hours per year, and no less than 2350 GPM for continuous pump operation. FPC did not consider (and still does not consider) this to be a reasonable pump flow limitation. Generic industry experience indicates that this value was overly conservative. FPC worked with Dresser to develop a recommended operating time limit for when a DH pump was operating with minimum flows less than the values they supplied. Dresser agreed that operation in the 80-100 GPM flow range was allowable for a maximum of 2 hours lifetime. However, Dresser stated that after 2 hours operating at 80-100 GPM, pump bearing and shaft maintenance would be necessary. FPC still did not believe that the limits associated with pump flow rates of 1200 GPM and 2350 GPM were reasonable values upon which to base future operation of CR-3. FPC decided to determine a more realistic time limit for low flow operation by actual pump testing.

FPC requested B&W Nuclear Service Company to determine which postulated accident could create the limiting minimum flow for the DH pumps. (B&W was selected based on their knowledge of the loss-of-coolant accident (LOCA) models). The B&W assessment was that a 0.007 ft^2 SBLOCA in the reactor coolant (RC) pump cold leg discharge line would be most representative of a small break LOCA that causes the RCS pressure to stabilize at a high value for a long time and create the most significant challenges to the operators in terms of their ability to cool and depressurize the plant in a controlled and timely manner. B&W estimated that the DH pump would be in a low flow condition (as low as 500 GPM) for approximately 5 hours. While FPC considered this time conservative, we nevertheless established a test on one of CR-3's DH pumps that required the pump to operate at 400 GPM for 10 hours. The test also included operating the pump with flows ranging from as low as 80 GPM to as high as 3000 GPM.

This test was conducted in April, 1989 and was witnessed by representatives of the NRC (Reference D discusses that test). Following the test, the DH pump was completely disassembled and inspected. No indications of damage to pump bearings, shaft, or rotating element were observed. However, the bearings were replaced before the pump was declared operable. Following the testing of this pump, and in consultation with MPR Associates, Inc., FPC established operating time limits in CR-3 operating procedures which require operators to monitor the time a pump is being operated in each flow regime. These DH pump flow limits and operating times are as follows:

- Indicated flow > 1400 GPM* — No time limit
- 300 GPM* ≤ indicated flow ≤ 1400 GPM* — 72 hrs maximum
- Indicated flow < 300 GPM* — 2 hrs maximum

* Pump flow is actually 100 GPM higher than these values since indicated flow does not include the 100 GPM pump bypass line flow.

In addition to establishing operating time limits at low flows, FPC evaluated the DH pump low flow performance by examining operating history, as well as additional technical evaluations. These evaluations, developed with the assistance of MPR Associates, Inc., were submitted by Reference B. FPC concluded from these evaluations that the operating history of these DH pumps does not indicate a connection between low flow operation and pump degradation.

FPC continued to have a dialogue with the NRC during the summer of 1989 concerning the acceptability and duration of DH pump operation in low flow conditions. This discussion culminated in a NRC/FPC/B&W meeting on August 17, 1989. As a result of this meeting (summarized in Reference C), the NRC asked FPC to develop a realistic DH pump operating time for a small break loss-of-coolant accident (SBLOCA) using a simulator. The results of that evaluation were submitted by Reference E. The results show that the low flow operation time period of the DH pump is less than the B&W LOCA model time of 5 hours and certainly less than the 10 hour test.

Discussions with the NRC have continued since the submittal of Reference D because the B&W simulator evaluation assumed that RC pumps were available throughout the accident. The NRC has felt that this was a non-conservative assumption given that classical licensing LOCA analyses also assume that the LOCA occurs with a loss-of-offsite power (LOOP). The NRC believes that a LOCA without a LOOP does not produce the most conservative mission time for a DH pump.

In an attempt to bring this issue to closure for CR-3, a NRC/FPC/B&W meeting (March 30, 1992) discussed whether FPC would perform an evaluation of the DH pump mission time using the CR-3 simulator or have B&W perform another evaluation. FPC determined that the CR-3 simulator would be the most effective way to do this evaluation.

The CR-3 simulator sequence is discussed below. The scenario assumptions used by FPC to evaluate this event were:

1. CR-3 is operating at 100% power, a SBLOCA occurs with a 0.007 ft^2 area,
2. A simultaneous LOOP occurs resulting in no RC pumps,
3. No single failure assumed,
4. 100% decay heat,
5. Emergency Core Cooling systems (ECCS) will operate at CR-3 setpoints and function as designed,
6. 2 high pressure injection (HPI) pumps and 2 low pressure injection (LPI) pumps available, and
7. Actions will be governed by existing CR-3 Abnormal, Emergency, and Operating Procedures.

These assumptions were discussed at the March 30, 1992 meeting and agreed to as discussed in Reference A.

CR-3 SIMULATOR SCENARIO DESCRIPTION

Table 1 provides a time line for the scenario. The following narrative description aids in fully understanding Table 1:

> The event began with a LOOP which resulted in a reactor trip. Concurrently, a SBLOCA developed in the "A" cold leg at the discharge of the "A" RC Pump. Reactor Coolant System (RCS) pressure degraded as the coolant discharged from the break. Adequate subcooling margin was lost as RCS pressure continued to decrease to the saturation line. HPI actuated at approximately 1500 psig. Full balanced HPI flow was delivered to the RCS from two HPI pumps. The "A" hot leg "candy cane" voided completely. Once Through Steam Generator (OTSG) level was raised to 95% with OTSG pressure controlled to maintain approximately 50°F difference between saturation temperature of the OTSGs and incore temperature. The RCS remained inadequately subcooled with RCS pressure decreasing until the LPI actuation setpoint of 500 psig was reached while both core flood tanks slowly discharged their contents to the RCS.

Upon LPI actuation, both DH Pumps started in the LPI mode. RCS pressure was above the shutoff head of the DHPs resulting in "mini-recirculation" flow of approximately 80 GPM. Procedural guidance for LPI flow below 300 GPM requires opening the PORV to reduce RCS pressure, potentially enabling LPI flow to increase. If opening the power-operated relief valve (PORV) does not achieve the desired effect, then placing both DHPs in a piggy back alignment from the Borated Water Storage Tank (BWST) to the suction of the HPI pumps was the next procedural option. This option proved to be successful. When the reactor building (RB) water level reached 2.2 feet, both DHPs were aligned to take suction from the RB sump and isolated from the BWST.

The RCS continued to depressurize as HPI/break cooling removed decay heat. HPI flow was throttled to prevent exceeding pump runout. Upon recovering adequate subcooling margin, HPI was further throttled to maintain less than 100°F subcooling. The cooldown and depressurization continued enabling DHP-1A to be realigned to provide decay heat removal. With the RCS water solid, pressure control was a function of throttling HPI flow. Upon reaching an incore temperature of ≤ 200°F, the simulation was terminated. The following actions would have been performed had the simulation continued:

1. Decrease RCS pressure by throttling HPI flow.
2. With RCS pressure ≤ 100 psig then shutdown DHP-1A.
3. Align the RCS drop line to the RB sump.

The time necessary to perform the above alignment is conservatively assumed to be 1000 seconds. DHP-1B would remain running with a flow rate less than 1500 gpm until the evolution was completed, therefore this additional time is included in the mission time.

DECAY HEAT PUMP MISSION TIMES

DHP-1A operated at a flow rate of < 400 GPM for approximately 10 minutes, while at shut off head. DHP-1A was required to operate at a flow rate of < 1,500 GPM for approximately 2 hours and 7 minutes, while in a piggy back mode.

DHP-1B operated at a flow rate of < 400 GPM for approximately 10 minutes, while at shut off head. DHP-1B was required to operate at a flow rate of < 1,500 GPM for approximately 5 hours and 15 minutes.

SUMMARY

The conclusions presented in previous FPC correspondence on the issue of DH pump operating times remain valid. The 0.007 ft^2 break is a limiting case for establishing DH pump mission times since it does cause a decoupling of the secondary and primary systems in 420 seconds. The SBLOCA scenario does take time to mitigate, but ECCS pump operation is not in jeopardy at any time. The 1989 DH pump test for 10 hours is more than adequate to demonstrate that the CR-3 pumps can perform their intended function. The results of the simulator evaluation serve to add credibility to the results of the original B&W SBLOCA analysis that concluded the event would require DH pump operation in a low flow condition for approximately 5 hours. FPC also continues to support the conclusions of the MPR report that the CR-3 DH pumps can support operation well beyond the 10 hours of actual pump test.

The results of this simulator evaluation and the previous efforts lead FPC to continue to conclude there is a adequate assurance that the CR-3 DH pumps can perform their intended functions for design basis accidents. This conclusion is based upon LOCA analysis which draw upon approved NRC models, results from actual pump testing, and two independent simulator evaluations. Bulletin 88-04 for the CR-3 DH pumps should be considered resolved.

TABLE 1

KEY EVENTS/EVOLUTIONS

Time (Sec)	Description of Event or Evolution
0	Loss of Offsite Power, Reactor Trip, SBLOCA.
125	HPI flow through two lines begin.
140	Loss of Adequate Subcooling Margin.
150	RCS Pressure reaches 1500 PSIG.
420	"A" Hot leg reaches 100% voiding.
490	Reactor building pressure at 4 PSIG.
4800	Core Flood Tanks begin to discharge.
5100	Throttled HPI flow to prevent pump runout.
6500	Low Pressure Injection Actuation - Both Decay heat pumps start in the LPI mode, but are operating on minimum recirculation
7100	Established piggyback mode of operation. Both trains of LPI aligned from BWST to HPI pumps.
7100	Throttled HPI to prevent pump runout.
11,600	Regained adequate subcooling margin.
12,000	Established suction for both LPI pumps from the RB sump. Maintained piggyback operation to HPI pumps.
13,100	"A" Hot leg reaches 0% voiding.
13,800	Began throttling HPI to maintain subcooling margin control band.
13,900	Established subcooling margin of ≥ 70°F.
14,700	RCS pressure ≤ 250 PSIG.
14,700	Shut down DHP-1A (LPI mode) to align for decay heat removal mode.
19,400	Established decay heat removal using DHP-1A.
24,000	RCS temperature ≤ 200°F (Incore).
25,000	Align the RCS drop line to the RB sump.