Enclosure 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 6, 1980

TO ALL OPERATING B&W REACTOR LICENSEES

As you know, on February 26, 1980, the Crystal River Unit No. 3 Nuclear Station (CR-3) experienced a reactor trip from approximately 100% full power. The initiating event was a failure in the power supplies for the non-nuclear instrumentation. A discussion of the event was presented by the Florida Power Corporation (FPC) in a meeting attended by representatives of your company in Bethesda, Maryland, on March 4, 1980. FPC also discussed the planned corrective action that would be taken at CR-3. The sequence of events presented by FPC and the planned corrective actions at CR-3 are attached to this letter as Enclosures 1 and 2 respectively.

Representatives from all other B&W operating plants were also present at the March 4, 1980 meeting. Each licensee addressed the history of non-nuclear instrumentation problems at his facility, the susceptibility of his plant(s) to the CR-3 event, and any corrective action that has been, or will be, taken.

On a related matter, Office of Inspection and Enforcement Bulletin 79-27 was issued subsequent to a similar event at the Oconee Nuclear Station, Unit No. 3, on November 10, 1979. This bulletin requested your review of certain matters relative to the Oconee event as they apply to your facility. Our interest in the CR-3 event, and its implication on the operation of your facility, does not relieve you of your responsiblities to provide the information requested by IE Bulletin 79-27.

Because of the implications of the CR-3 event, and potential adverse effects on the public health and safety that could result from future events of this type, we believe that certain information in addition to that requested in IE Bulletin 79-27, should be promptly provided to the NRC concerning your facility. In accordance with 10 CFR 50.54 (f), you are requested to provide us with information in response to Items 1 through 5 of Enclosure 3, submitted under oath or affirmation, no later than close of business March 12, 1980. Information in response to Items 6 and 7 of Enclosure 3 should be submitted no later than close of business March 17, 1980. The information provided in your responses will enable us to determine whether or not your license should be modified, suspended, or revoked.

This letter confirms the oral request for this information expressed at the March 4, 1980 meeting by Mr. Darrell G. Eisenhut, Acting Director, Division of Operating Reactors.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures: As stated

SEQUENCE (AS OF 2300 3/1/80)

26 February Transient CR-3

EVENT SYNOPSIS

At 14:23 on February 26, 1980 Crystal River -3 Nuclear Station experienced a reactor trip from approximately 100% full power. A synopsis of key events and parameters was obtained from the plant computer's post-trip review and plant alarm summary, the sequence of events monitor, control room strip charts, and the Shift Supervisor's log.

The reactor was operating at approximately 100 % full power with Integrated Control System (ICS) in automatic. No tests were in progress and minor maintenance was being performed in the Non-Nuclear Instrumentation (NNI) cabinet "Y".

Time	Event	Cause/Comments
14:23:00	The following is a summary of plant conditions prior to the trip Flux 98.6% RC Pressure 2157 psig PZR level 202 inches MU tank level 71 inches T_H "A" 599°F. T_H "B" 600°F. T_C "A" 557°F. T_C "B" 556°F. RC Flow "A" 73 X 10^6 lbs/hr RC Flow "B" 73 X 10^6 lbs/hr Letdown Flow 48 gpm OTSG "A" lvl (OP) 67% OTSG "B" lvl (OP) 65% OTSG "A" FRLV 242 inches OTSG "B" FRLV 254 inches OTSG "A" Pressure 911 psig OTSG "B" pressure 909 psig Main Steam Pressure 894 psig Main Steam Temp. 589°F. Condenser Vacuum 1.76 Generated MW 834 DFT level 12.7 ft. Feed Flow "A" 5 X 10^6 lbs/hr Feed Flow "B" 5 X 10^6 lbs/hr Feed Pressure "A" 970 psig Feed Pressure "B" 968 psig	
14:23:21	+24 Volt Bus Failure (NNI power loss "X" supply)	Cause still unknown. Apparently, the positive 24 VDC bus shorted dragging the bus voltage down to a

Time	Event	Cause/Comments
		low voltage trip condition. There is a built-in ¼ to ½ second delay at which time all power supplies will trip. There was no trip indication on negative (-) voltage. This event was missed by the annunciator. Following the NNI power failure, much of the control room indication was lost. Of the instrumentation that remained operable transient conditions made their indication questionable to the operators.
14:23:21	PORV and Spray Open	When the positive 24 VDC supply was lost due to the sequence discussed above the signal monitors in NNI changed state causing PORV/Spray valves to open. The PORV circuitry is designed to seal in upon actuation and did so. The resultant loss of the negative 24 VDC halted spray valve motor operator and prevented PORV seal in from clearing on low pressure. It is postulated that the PORV opened fully and the spray valve stroked for approximately ½ second. The 40% open indication on spray valve did not actuate, therefore, the spray valve did not exceed 40% open.
14:23:21	Reduction in Feedwater	As a result of the "X" power supply failure many primary plant control signals responded erroneously. Tcold failed to 570°F (normal indication was 557°F) producing several spurious alarms. Tave failed to 570°F (decreased). The resultant Tave error modified the reactor demand such that control rods were withdrawn to increse Tave and reactor power. The power increase was terminated at 103% by the ICS and a "Reactor Demand High Limit" alarm was received. Thot failed to 570°F (low) and RC flow failed to 40 X 10^6 lbs/hr in each loop (low). Both these failures created a BTU alarm and limit on feedwater which reduced feedwater flow to both OTSG's to essentially zero. Turbine Header Pressure failed to 900 psig (high) which caused the turbine valves to open slightly to

Time	Event	Cause/Comments
		regulate header pressure thus increasing generated megawatts. These combined failures resulted in a loss of heat sink to the reactor initiating an excessively high RC pressure condition.
14:23:35	Reactor Trip/Turbine Trip	Rx trip caused by high RCS pressure at 2300 psi Turbine was tripped by the reactor.
14:24:02	Hi Pressure Inj. Req. (Flag)	This was a computer printout and indicates <50° subcooling.* See attached graph of RC Pressure/Temp. vs. Time. This graph is based on Post Trip data and actual incore thermocouple data. From the reactor trip point (14:23 to 14:33, core exit temperature data was obtained by extrapolation and calculated data. This is supported by two alarm data points plotted at 18° and 21° of subcooling during this period from the computer. It is important to note that lowest level of subcooling was 8°F for a very short period of time. *NOTE: This computer program was initiated as a result of the TMI incident.
14:24:02	Loss Of Both Condensate Pumps	Suspect condensate pump tripped due to high DFT level. This is verified by ???? printed by computer, indicating the level instrument was over ranged as well as a low flow indication in the gland steam condenser as also indicated by computer.
14:25:50	PORV Isolated	At this time a high RC Drain Tank level alarm was received. This was resultant from the PORV remaining open and was positive indication that the PORV was open. At this time, the operator closed the PORV block valve due to RCS pressure decreasing and high RCDT level.
14:26:41	HPI Auto Initiation	HPI initiated automatically due to low RCS pressure of 1500 psig. The low pressure condition was resultant from the PORV remaining full open while the plant was tripped. Full HPI was initiated with 3 pumps resulting in approximately 1100 gpm flow to the RCS. At this time, all remaining non-essential R.B. isolation valves

Time	Event	Cause/Comments
		were closed per TMI Lessons Learned Guidelines.
14:26:54	RC Pumps Shutdown	Operator turned RC pumps off as required by the applicable emergency procedure and B & W small break guidelines.
14:27:20	RB Pressure Increasing	This is first indication that RCDT rupture disc had ruptured. RB pressure increase data was obtained from Post Trip Review and Strip Chart indication.
14:31:32	RB Pressure High	This alarm was initiated by 2 psig in RB. This is attributed to steam release from RCDT. Code safeties had not opened at this time based upon tail pipe temperatures recorded at 14:32:03 (Computer).
14:31:49	OSTG "A" Rupture Matrix Actuation	This occurred due to <600 psig in OTSG "A. The low pressure was caused by OTSG "A" boiling dry which was resultant from the BTU limit and failed OTSG level transmitter. This resulted in the closure of all feedwater and steam block valves which service OTSG "A".
14:31:59	Main Feedwater Pump 1A Tripped	Caused by suction valve shutting due to matrix actuation in previous step.
14:32→14:41	ES A/B Bypass	Manually bypassed and HPI balanced between all 4 nozzles (Total flow approximately 1100 gpm -small break operating guidelines).
14:32:35	Started Steam Driven Emergency Feedwater Pump	Started by operator to ensure feedwater was available to feed OTSG's.
14:33	Core Exit Temp. Verified	The core exit incore thermocouples indicated the highest core outlet temperature value was 560°F. RCS pressure was 2353 psig atthis time, therefore, the subcooling margin at this time was 100°F. Minimum subcooling margin for the entire transient was 8°F. It is postulated that some localized boiling occurred in the core at this point as indicated by the self powered neutron detectors.
14:33→14:44	Started Motor Driven Emergency Feedwater Pump	Same discussion as "Started Steam Driven Emergency Feedwater Pump."
14:33:30	RC Pressure High (2395 psig)	At this point, pressurizer is solid and code safety lifts (RCV-8). This is the highest RCS pressure as recorded on Post Trip Review. Apparently, RCV-8 lifted early due to seat

Time	Event	Cause/Comments
		leakage prior to the transient and RCV-9 did not lift.
14:34:23	RB Dome Hi Rad Level	RMG-19 alarmed at this point. Highest level indicated during course of incident was 50 R/hr. High radiation levels in RB caused by release of non-condensable gases in the pressurizer and coolant.
14:35:33	Attempted NNI Repower Without Success	This resulted in spikes observed on de-energized strip charts.
14:36:50	Computer Overload	Caused by overload of buffer. Resulting in no further computer data until buffer catches up with printout.
14:38:15	FWV-34 Closed	This valve was closed to prevent overfeeding OTSG "B" beyond 100% indicated Operating Range.
14:44:12	NNI Power Restored Successfully	NNI was restored by removing the X-NNI Power Supply Monitor Module. This allowed the breakers to be reclosed. At this time, it was observed that the "A" OTSG was dry, the pressurizer was solid (Indicated off scale high), RC outlet temperature indicated 556°F (Loop A & B average), and RC average temperature indicated 532°F (Loop A & B). The highest core exit thermocouple temperature at this time was 531°F RSC pressure was 2400 psig (saturation temp. at this pressure is 662°F.). This data verified natural circulation was in progress and the plant subcooling margin was 131°F. (based on core exit thermocouples).
14:44:31	RB Isolation and Cooling Actuation	At this time, RB pressure increased to 4 psig and initiated RB Isolation. The operator verified all immediate actions occurred properly for HPI, LPI, and RB Isolation and Cooling. The increasing RB pressure was resultant from RCV-8 passing HPI at this time.
14:46:10	Bypassed HPI, LPI and RB Isolation and Cooling	These "ES" systems were bypassed at this time to again balance HPI flow and restore cooling water to essential auxiliary equipment (i.e., RCP's, letdown coolers, CRDM's etc.).

Time	Event	Cause/Comments
14:51:57	Rupture Matrix Actuation on OTSG-B	The actuation was resultant from a degradation of OTSG-B pressure. Cold emergency feed was being injected into the OTSG at this time. This matrix actuation isolated all feedwater and steam block valves to the B-OTSG and tripped the "B" main FW pump. Both Emergency FW pumps were already in operation at this time. B-OTSG level at this time was 70% (Operation Range).
14:52	HPI Throttled and RCS Pressure Reduced to 2300 psig	At this time, the maximum core exit thermocouple temperature was 515°F, RCS pressure was 2390 psig. Therefore, the subcooling margin was 147°F. Natural circulation was in effect as verified previously. All conditions had been satisfied to throttle HPI. Therefore, flow was throttled down to approximately 250 gpm to reduce RCS pressure to 2300 psig in order to attempt to reduce the flow rate through RCV-8 and into the RB.
14:53	Reestablished Letdown	At this time, the operator was attempting to establish RCS pressure control via normal RC makeup and letdown.
14:56	Opened MU Pump Recirc. Valves	This was done to assure the MU pumps would have minimum flow at all times to prevent possible pump damage.
14:56:43	Bypassed the A-OTSG Rupture Matrix and Reestablished Feed to the A-OTSG	Feedwater was slowly admitted to the A-OTSG which was dry up to this point. Feedwater was admitted through the Auxiliary FW header via the EFW bypass valves. The feedrate was very slow in order to minimize thermal shock to the OTSG and resultant depressurization of the RCS. RCS pressure control was very unstable at this time. It is postulated that some localized boiling occured in core at this point as indicated by self neutron detectors.

Time	Event	Cause/Comments
14:57:09	Bypassed the B-OTSG Rupture Matrix	This was done to regain FW control of the B-OTSG. Level was still high in this OTSG (approximately 65% Operating Range). Therefore, feed was not necessary at this time. The Main Steam Isolation valves were open in preparation for bypass valve operation (when necessary).
14:57:15	Established RC Pump Seal Return	This was done in preparation for a RCP start (when necessary) and to minimize pump seal degradation.
15:00:09	Reestablished Level In A-OTSG	This verified feedwater was being admitted to the OTSG and made it available for core cooling via natural circulation. Feed to this generator was continued with the intent of proceeding to 95% on the Operating Range.
15:00:09	77°F Subcooled "A" Loop	This value was based upon "A" RCS loop parameters at this time. The "A" loop was being cooled down at this time by the A-OTSG fill and the operator was attempting to equalize loop temperatures.
15:15	23°F Delta-T/Manned the Technical Support Center	At this time, loop temperatures were nearing equalization. This delta-T was calculated from loop A & B T_c's and core exit thermocouples.
15:17	Declared Class "B" Emergency	This was done based on the fact there was a loss of coolant through RCV-8 in the containment and HPI had been initiated. All non-essential CRF 3 personnel were directed to evacuate and contact off-site agencies began. Survey team was sent to Auxiliary Building
15:19	Opened Emergency FW Block to B-OTSG	At this point the A-OTSG level was increasing and the decision was made to commence filling the B-OTSG simultaneously. The intent was to go 95% on both OTSG's without exceeding RCS cooldown limits (100°F/hr) while maintaining RCS pressure control.

Time	Event	Cause/Comments
15:26	Lo Level Alarm in Sodium Hydroxide Tank	This was recultant from the tank supply valve opening when the 4 psig RB isolation and cooling signal actuated. The sodium hydroxide was released to both LPI trains. Sodium Hydroxide was admitted to the RCS via HPI from the BWST. (Approximately 2 ppm injected into the RCS.)
15:50	Terminated HPI	At this time, all conditions had been satisfied (per small break operating guidelines) to terminate HPI. RCS pressure control had been established using normal makeup and letdown. HPI was terminiated and essentially all releases to the RB were discontinued.
16:00	Commenced Pressurizer Heatup	At this time, RCS pressure and temperature were well under control. Natural circulation was functioning as designed (approximately 23°F delta-T). RCS temperature was being maintained at approximately 450°. RCS pressure was approximately 2300 psig. The decision was made at this point to commence pressurizer heatup in preparation to re-establish a steam space in the pressurizer.
16:07	Survey Team Report	The Emergency Survey Team reported no radiation survey results taken offsite were above background.
16:08 :04	Shutdown Steam Drive Emergency FW Pump	The motor driven Emergency FW pump was running, therefore, the steam driven pump was not needed The plant remained in this condition for approximately 2 hours, while heating up the pressurizer to saturation temperature for 1800 psig.
16:15	Press Release	Media was notified of plant status.
18:05	Established Steam Space Pressurizer	At this point, pressurizer temperature was approximately 620°F. Pressurizer level was brought back on scale by increasing letdown. From this point pressurizer level was reduced to normal operating level and normal pressure was established via pressure heaters.
18:30	Terminated Class B Emergency	State and Federal Agencies notified.

Time	Event	Cause/Comments
21:07	Forced Flow Initiated in RCS	The decision was made to re-establish forced flow cooling in the RCS at this time. B&W and NRC were consulted. RCP-1B and 1D were started. At this point, RCS parameters were stabilized and maintained at RC pressure-2000 psig, RCS temperature-420°F. Pressurizer level-235 inches. The plant was considered in a normal configuration.

PLANNED CORRECTIVE ACTION AT CR-3

Immediate

- Thorough testing of NNI system to determine cause of failure
- Modify PORV so that NNI failure closes valve
- Modify pressurizer spray valve so that valve doesn't open on NNI failure
- Provide positive indication of all three relief or safety valves
- Establish procedural control of NNI Selector switches
- Train all operators in response to NNI failures
- Move 120v ICS "X" power to vital bus
- Initiate more extensive program for events recorder system
- Provide operator with redundant indication of main plant parameters

At Next Refueling (September 1980)

- Install indication lights on all panels to know if power on panel
- Quick access to fuses is being designed into cabinets
- Modify EFW pump circuit to start pumps on any low steam generator level signal

Long Term

- Investigate upgrade of NNI capabilities - total loss of NNI
- Remote shutdown is being designed
- Provide backup AC sources to inverters with automatic transfer.

Information requested by COB, March 12 and March 17, 1980.

1. Summarize power upset events on NNI/ICS that have previously occurred at your plant.

2. Specifically review the Crystal River event, and address your plant's susceptibility to it in general.

3. Set forth the information presented by your representative(s) in the meeting on March 4, 1980.

4. Address information available to the operator following various NNI/ICS power upset events, including a discussion of:

   - how the operator determines which information is reliable

   - what information is needed to bring the plant to cold shutdown

5. Address the feasibility of performing a test to verify reliable information that remains following various NNI/ICS power upsets.

6. Address each CR-3 proposed corrective action in terms of applicability to your plant.

7. Expand your review under IE Bulletin 79-27 to include the implications of the CR-3 event. Inform us of your schedule for completion of this expanded review as discussed on March 4, 1980.

In addition to the above, Florida Power Corporation should address:

1. Sequence of events for the CR-3 trip.

2. Proposed corrective actions at CR-3.

3. Discuss the impact, whether it be beneficial or detrimental, of NRC Short Term Lessons Learned and Bulletins and Orders requirements.