ENCLOSURE 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

IE Information Notice No. 80-10
Date: March 7, 1980

PARTIAL LOSS OF NON-NUCLEAR INSTRUMENT SYSTEM POWER SUPPLY DURING OPERATION

Description of Circumstances:

This notice contains information regarding Crystal River Unit 3 response to a loss of non-nuclear instrumentation (NNI) as a consequence of loss of the +24 volt power supply to the NNI.

At 2:23 p.m. on February 26 with Crystal River Unit 3 at 100% power, the +24 volt power supply to the NNI was lost, due to a short to ground. This initiated a sequence of events (detailed in the enclosure) wherein the PORV opened and stayed open as a direct result of the NNI power supply loss. HPI initiated as a result of depressurization through the open PORV, and with approximately 70% of NNI inoperable or inaccurate, the operator correctly decided that there was insufficient information available to justify terminating HPI. Therefore, the pressurizer was pumped solid, one safety valve lifted, and flow through the safety valve was sufficient to rupture the RC Drain Tank rupture disk, spilling approximately forty-three thousand gallons of primary water into containment.

The Crystal River 3 event is closely related to the November 10, 1979 event at Oconee Unit 3 wherein the inverter supplying power to the Integrated Control System (ICS) and to parts of the NNI failed. That event was the subject of IE Information Notice 79-29 (November 16, 1979) which was followed by IE Bulletin 79-27 (November 30, 1979).

The CR-3 event involved loss of only part of the power available from an inverter, rather than the inverter itself, since the +24v supply is only one of several power supplies drawing power from one inverter. The effects are very similar, however, in that the ICS lost part of its input signals in both events.

The +24 volt power supply short to ground has tentatively been identified by the licensee to have occurred between knife edge connectors of a Bailey Control Company Voltage Buffer Card. The voltage buffer card was misaligned in its receptacle, and adjacent connectors carrying +24v and "common" were bent such that they contacted one another. This short circuit cleared itself during subsequent re-energizing of the power supply by burning through the foil on a printed circuit card. Subsequent review by the licensee identified a second voltage buffer card which was also misaligned but had not caused a short circuit.

The specific circuit cards which were misaligned carried part number 6624609L1. The connectors on these cards are slightly thinner and appear to have a somewhat different angle than those found on similar cards elsewhere in the NNI which carry part numbers 6624608A1 or 6624609A1. The 6624609LI cards appear to be more subject to misalignment.
The specific shorted voltage buffer card provided the signal to the NNI "x" saturation meter.

Licensees which utilize Bailey Control Company Voltage Buffer Cards are requested to carefully inspect the cards for possible misalignment and take corrective actions if misalignments are identified. Specific instructions for carrying out these inspections and providing any other information which may be required to define appropriate corrective action is being prepared by Baily Control Company for transmittal to purchasers of this equipment by March 11, 1980.

Initial screening of IE Bulletin No. 79-27 responses indicates a range of responses regarding depth and scope of review.

IE Bulletin No. 79-27 was intended to cause licensees to investigate loss of individual power supplies as well as total loss of an inverter or vital bus. An addendum to IE Bulletin No. 79-27 is planned to be issued in the near future to reflect the CR-3 event.

This Information Notice is provided to inform licensees of a possibly significant matter. It is expected that recipients will disseminate the information to all operational personnel working at their licensed facilities. (A meeting was held with B&W licensees in NRC Headquarters on March 5, 1980 to review the event at Crystal River and to discuss proposed corrective actions. Responses to specific questions have been requested of the B&W licensees.) If you have questions regarding this matter, please contact the Director of the appropriate NRC Regional Office.

No written response to this Information Notice is required.

Enclosure:
Sequence of Events

26 FEBRUARY TRANSIENT CRYSTAL RIVER UNIT-3
SEQUENCE OF EVENTS

EVENT SYNOPSIS

At 14:23 on February 26, 1980, Crystal River-3 Nuclear Station experienced a reactor trip. Nominal full power primary and secondary system parameters were present. 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.

Time	Event	Cause/Comments
14:23:21	+24 Volt Bus Failure (NNI power loss "X" supply)	The positive 24 VDC bus shorted dragging the bus voltage down to a low voltage trip condition. There is a built-in 1/4 to 1/2 second delay at which time all power supplies will trip. The trip indication on negative (-) voltage 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 indications questionable to the operators.
14:23:21	PORV and Pressurizer Spray Valve 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 spral valve stroked for approximately 1/2 second. The "40% open" indication on spray valve did not actuate, therefore, the spray valve did not exceed 40% open.

Time	Event	Cause/Comments
14:23:21	Reduction in Feedwater	As a result of the "X" power supply failure many primary plant control signals responded erroneously. T-cold failed to 570°F (normal indication was 557°F) producing several spurious alarms. T-ave failed to 570°F (decreased). The resultant T-ave error modified the reactor demand such that control rods were withdrawn to increase T-ave and reactor power. The power increase was terminated at 103% by the ICS and a "Reactor Demand High Limit" alarm was received T-hot 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 feed-water 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 regulate header pressure thus increasing generated megawatts. These combined failures resulted in a loss of heat sink to the reactor initiating high RCS 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. Reg. (Flag)	This was a computer printout and indicates < 50° subcooling. (The lowest level of subcooling was 8°F for a very short period of time, at about 14:30)
14:24:02	Loss of Both Condensate Pumps	Suspect that the condensate pump tripped due to high De-aeriating feed tank level.

Time	Event	Cause/Comments
14:24:02 (Continued)		This is verified by a series of questions marks ???? printed by computer indicating that the level instrument was over-ranged. A low flow indication in the gland steam condenser was 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 reactor building (RB) isolation valves 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.

Time	Event	Cause/Comments
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 power supply to OTSG "A" 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 rupture 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 at this time, therefore, the subcooling margin at this time was <u>100°F</u>. Minimum subcooling margin for the entire transient was <u>8°F</u> at 14:30. It is postulated that some localized boiling occurred in the core at this point as indicated by the self powered neutron detectors.

Time	Event	Cause/Comments
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 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. Resulted 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

Time	Event	Cause/Comments
14:44:12 (Continued)		this time was 531°F. RCS 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 relieving pressure due to continued HPI.
14:46:10	Bypassed HPI, LPI and RB Isolation and Cooling	These "ES" systems were bypassed at this time to balance HPI flow and restore cooling water to essential auxiliary equipment (i.e., RCP's, letdown coolers, CRDM's etc.).
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.

Time	Event	Cause/Comments
14:52 (Continued)		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 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.
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).

Time	Event	Cause/Comments
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	Reëstablished 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 into the containment and HPI had been initiated. All non-essential CR-3 personnel were directed to evacuate and contact of 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 resultant 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 terminated 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°F. 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 Driven 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.

Time	Event	Cause/Comments
18:05	Established Steam Space in 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 pressurizer heaters.
18:30	Terminated Class B Emergency	State and Federal Agencies notified.
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.

ENCLOSURE 2

RECENTLY ISSUED IE INFORMATION NOTICES

Information Notice No.	Subject	Date Issued	Issued to
79-35	Control of Maintenance and Essential Equipment	12/31/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-36	Computer Code Defect in Stress Analysis of Piping Elbow	12/31/79	All Power Reactor Facilities with an OL or CP
79-37	Cracking in Low Presssure Turbine Discs	12/31/79	All Power Reactor Facilities with an OL or CP
80-01	Fuel Handling Events	1/4/80	All Power Reactor Facilities with an OL or CP
80-02	8X8R Water Rod Lower End Plug Wear	1/25/80	All BWR Facilities with an OL or CP
80-03	Main Turbine Electro-hydraulic Control System	1/31/80	All Power Reactor Facilities with an OL or CP
80-04	BWR Fuel Exposure in Excess of Limits	2/4/80	All BWR Facilities with an OL or CP
80-05	Chloride Contamination of Safety Related Piping and Components	2/8/80	All Power Reactor Facilities with an OL or CP and applicants for a CP
80-06	Notification of Significant Events	2/27/80	All Power Reactor Facilities with an OL and applicant for OL
80-07	Pump Fatigue Cracking	2/29/80	All Power Reactor Facilities with an OL or CP
80-08	The States Company Sliding Link Electrical Terminal Block	3/7/80	All Power Reactor Facilities with an OL or CP
80-09	Possible Occupation Health Hazard Associated with Closed Cooling Systems for Operating Power Plants	3/7/80	All Power Reactor Facilities with an OL or CP