Indian Point 3
Nuclear Power Plant
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NewYorkPower Authority

Robert J. Barrett
Site Executive Officer

October 26, 1999
IPN-99-115

U.S. Nuclear Regulatory Commission
ATTN: Document Control Center
Washington, D.C. 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Licensee Event Report 1999-12-00
A Common Condition Causing Multiple Core Exit Thermocouples to be Inoperable During Postulated Accident Conditions Due to Moisture Intrusion.

Dear Sir:

The attached Licensee Event Report (LER) 1999-12-00 is submitted as required by 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73(a)(2)(vii).

There are no commitments being made in this LER.

Very truly yours,

Robert J. Barrett
Site Executive Officer
Indian Point 3 Nuclear Power Plant

Attachment

cc: see next page

cc: Mr. Hubert J. Miller
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INPO Record Center
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Atlanta, Georgia 30339-5957

U. S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

NRC FORM 366 (6-1998)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)
A Common Condition Causing Multiple Core Exit Thermocouples to be Inoperable During Postulated Accident Conditions Due to Moisture Intrusion

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
09	30	1999	1999	-- 12 --	00	10	26	1999	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	000		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)	✓	50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)	
NAME	TELEPHONE NUMBER (Include Area Code)
Richard Burroni, Instrumentation & Controls Manager	(914) 736-8794

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	IP	TE	CKB Industries	N					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	✓	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On September 30, 1999 the unit was in cold shutdown due to a refueling outage and the core was off loaded to the spent fuel pool. It was determined that ten (10) CKB Industries safety-related core exit thermocouples would be inoperable during post accident conditions due to moisture intrusion. The exact time of the moisture intrusion condition for the thermocouples is unknown and may have occurred between the previous refueling outage and September 30, 1999. A meggar insulation resistance (IR) measurement on all Regulatory Guide 1.97 qualified thermocouples indicated that these ten thermocouples failed to meet the IR requirements. These ten (10) Regulatory Guide 1.97 thermocouples were replaced. The replacement core exit thermocouples are not manufactured by CKB Industries. This event had no effect on the health and safety of the public.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Description of Event

In Westinghouse Energy Systems Business Unit's Nuclear Safety Advisory Letter (NSAL) 95-006, revision 1, it was identified that in-core thermocouples manufactured by CKB Industries have exhibited moisture intrusion. This moisture intrusion was detected by insulation resistance (IR) measurements performed following hydro and hot functional testing by Westinghouse. A significant number failed to meet the IR criteria established. Subsequent examination determined that the very low IR readings were caused by leakage through the weld in the tip area.

On September 30, 1999, the unit was in cold shutdown due to a refueling outage and the core was off loaded to the spent fuel pool. In response to the concerns expressed in NSAL-95-006, revision 1, dated October 19, 1995 all core exit thermocouples were tested in order to obtain insulation resistance (IR) readings. Testing revealed the IR values were below acceptable limits for ten (10) qualified thermocouples. All of these thermocouples would be inoperable during post accident conditions due to moisture intrusion. Seven of the ten thermocouples were replaced last outage for moisture intrusion. The other three thermocouples were upgraded to RG 1.97 qualified in the last refueling outage. The exact time of the moisture intrusion condition for the thermocouples is unknown and may have occurred between Refuel Outage 9 startup (September 12, 1997) and September, 30, 1999.

The Westinghouse IR criteria for acceptance of a thermocouple during the manufacturing process is typically 10E9 ohms or higher. Since thermocouples will operate properly under normal conditions with extremely low IR, it may not be practical to reject all thermocouples below 10E9 ohms. However, to address post accident performance based on leakage in the immersed tip area, Westinghouse has developed criteria to identify the source of the moisture intrusion. Based on the recent evaluation performed on thermocouples returned from other utilities, it was evident that leakage in the tip area would significantly reduce the IR, so a value of 10E6 ohms was selected as the threshold where it could be assumed that the IR degradation was caused by leakage in the wetted area of the thermocouple and most likely the tip weld.

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Presently, there are 20 qualified and 35 non qualified thermocouples in use at Indian Point 3. The difference between qualified and non qualified thermocouples is not the thermocouple itself, but the type of connector at the reactor head and the wiring up to the "bed spring" from the containment penetrations. Ten (10), non-qualified, thermocouples are currently out of service.

Low voltage thermocouple systems can tolerate low IR and still perform acceptably. The primary concern is the integrity of the thermocouple when subjected to rapidly increasing post-accident temperatures and decreasing post-accident pressures.

Cause of Event

This event was caused by moisture intrusion in the core thermocouples due to leakage at the immersed tip area. This can cause core exit thermocouple failures in a post-accident condition. The thermocouples that did not meet the criteria of NSAL 95-006, revision 1 for post-accident conditions did meet the criteria for operation under normal plant operating conditions.

Corrective Action

The following corrective actions have been performed to address this event:

- Performed a meggar, insulation resistance measurement, on all twenty (20) Regulatory Guide (RG) 1.97 qualified thermocouples. Testing revealed that ten (10) RG 1.97 thermocouples had low IR readings and failed to meet the IR requirements.

- Removed the ten (10) thermocouples that had low IR readings and replaced them with thermocouples that are manufactured by Imaging and Sensing Technology. Testing was performed on these thermocouples by the manufacturer. Testing that was done included radiography of the end cap welds and IR measurements taken at room temperature, elevated temperatures, after thermal cycling, and post hydro testing. These replacement thermocouples were fitted with qualified Conax connectors to meet the requirements of RG 1.97.

- Post-installation insulation resistance measurement testing was satisfactorily performed on October 2, 1999.

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Analysis of Event

This event is reportable under 10 CFR 50.73(a)(2)(vii). The licensee shall report any common cause or condition resulting in independent trains or channels becoming inoperable. Ten (10) RG 1.97 core-exit thermocouples may not have met the post-accident requirements from after startup from the last refueling outage to present due to the possibility of their not operating as designed during post-accident conditions.

Based on Westinghouse limited testing as described in their NSAL 95-006, revision 1, even if the thermocouples burst, they may still provide an adequate measurement. This potentially degraded condition my have degraded the digital subcooling margin monitor which uses the core-exit thermocouples as an input and is also used for emergency operating procedures. This was due to the possibility that these core exit thermocouples may not have operated as designed during the postulated accident condition.

A review of the past two years of Licensee Event Reports (LER) indicates that LER 97-012-00 indicates a similar condition occurred where a manufacturer defect rendered multiple trains or channels inoperable. This LER was for multiple core exit thermocouples to be inoperable during postulated accident conditions due to moisture intrusion.

Safety Significance

The core-exit thermocouples are not subject to the condition discussed in this event during normal plant operations and would have performed all their design functions for past plant operations. Therefore, there was no effect on the health and safety of the public for actual past plant operations. It is believed that for the postulated accidents causing the conditions discussed in this event that there is no significant effect on the health and safety of the public based on the following information/analysis from the Westinghouse NSAL 95-006, Revision 1:

Issues with leakage in the tip area of the in-core thermocouples are corrosion and bursting. Corrosion of the lead wires requires sufficient exposure time to temperatures above 500^{0}C and therefore should not be a problem at either normal or accident conditions. Although postulated accident temperature may exceed 500^{0}C, such temperature would exist for a limited time. Other sources indicate

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that under these conditions the corrosion depth is limited to 5 mils (lead wire diameter is 20 mils). Westinghouse also evaluated bursting of the thermocouple sheath due to rapidly increasing temperatures and decreasing pressures during post-accident conditions. This caused "flashing" of the trapped moisture. Three thermocouples with low IR were subjected to an extreme test where the tip was exposed to an instantaneous change from room temperature to 2000^0F. Two of the three burst but did not break the lead wire. Even if the thermocouples burst, they may still provide an adequate measurement.

The conditions under which the thermocouples could fail would only occur during a severe core heatup (above 1000^0F). Typically, there are only two accidents that would result in such high core temperatures - an inadequate core cooling (ICC) scenario (which is beyond the design basis of the plant) and the design basis small Loss of Coolant Accident (LOCA) scenario. The ICC scenario is a loss of coolant scenario for which there is no makeup to the primary system.

As the core heats up, the operator will perform recovery actions to restore Safety Injection (SI) flow and dump steam to reduce Reactor Coolant System (RCS) pressure (which can result in accumulator injection). Also, the operator may try to start Reactor Coolant Pumps (RCPs) to provide forced cooling in the core and to open a pressurizer Power Operated Relief Valve (PORV) to further depressurize the RCS. If any of these actions are successful in restoring core cooling, the operator will return to performing the normal recovery actions. If none of the actions are successful, the operator will eventually transition to the Severe Accident Management Guidelines to mitigate fission products that are released from the overheated core.

Note: Severe Accident Management enhancements were implemented at Indian Point Unit 3 on December 31, 1998 as previously committed in NYPA Letter IPN-95-040, dated March 28, 1995.

If during the core heatup some of the thermocouples fail, the operator should still have adequate indication from the remaining core exit thermocouples that the actions are either successful or have failed in restoring core cooling. Since the hotter thermocouples will fail first, the operator may not have the indication of the hottest core temperature. However, the downward trend of the core exit thermocouples should be adequate in determining the success of the recovery strategies.

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If all of the thermocouples fail during the heatup, the operator will not have an indication as to whether the recovery actions have successfully restored core cooling. Note that the maximum temperature expected during the design basis small break LOCA would be in the 1200 to 1300 ^{0}F range for a very short period of time (less than a few minutes). Therefore the operator may continue to perform recovery actions needlessly. Although these recovery actions are not detrimental to the safety of the plant, they could result in needless damage to plant equipment. An example would be starting an RCP in highly voided conditions during the worst point of the small break LOCA, which could destroy the pump.

A failure of some of the core exit thermocouples during an accident with high core temperatures will not jeopardize plant safety. Although the complete failure of thermocouples will not jeopardize plant safety for the design basis small break LOCA or the ICC scenario, it would complicate the recovery and could result in unnecessary damage to plant equipment.

Power Authority's Alternate Equipment Available:

Based on Westinghouse limited testing as described in NSAL 95-006, revision 1, even if the thermocouples burst, they may still provide an adequate measurement. In addition, the Reactor Vessel Level Indicating System (RVLIS) provides a means to monitor the water level in the reactor vessel during a postulated accident, although secondary in use to the thermocouples in the emergency operating procedures. It is designed to function under all normal, abnormal, accident and post-accident conditions concurrent with seismic events. The RVLIS consists of two redundant trains, with redundant power supplies, which automatically compensate for variation in fluid density as well as for the effects of reactor coolant pump operation. This system was installed in response to NUREG-0737, Item II.F.2 and RG 1.97 as a diverse means to detect inadequate core cooling. In accordance with the Technical Specifications RVLIS was operable from September 12, 1997 through September 10, 1999.

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Nuclear Power Plant
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NewYorkPower Authority

Robert J. Barrett
Site Executive Officer

October 26, 1999
IPN-99-116

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Withdrawal of Relief Request for Reactor Vessel Nozzle Inspections

Reference: 1. NYPA letter to the NRC, "Relief Request for Reactor Vessel Nozzle Inspections," IPN-99-088, dated August 18, 1999.

Dear Sir:

The purpose of this letter is to withdraw the relief request submitted in Reference 1. In Reference 1, the Authority requested the NRC to grant relief from the inspection requirements of ASME Code Section XI, for the volumetric examination of the inner radius section of the reactor vessel nozzles. This inspection was required to be performed during refueling outage RO 10.

The Authority has performed the required inspection during RO 10 and the relief is no longer needed. The next inspection is required to be performed during the next (third) 10-year inservice inspection interval and the Authority will resubmit the relief request, if required, as part of the Inservice Inspection Program submittal for the third 10-year interval. Therefore, the Authority hereby withdraws the relief request submitted in Reference 1.

The Authority is making no new commitments in this submittal. Should you have any questions regarding this matter, please contact Mr. K. Peters at (914) 736-8029.

Very truly yours,

Robert J. Barrett
Site Executive Officer

cc: see next page

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
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