Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511

914-736-8000

New York Power Authority

December 24, 1992
IP3-NRC-92-100

Docket No. 50-286
License No. DPR-64

Document Control Desk
Mail Station PI-137
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

The attached Licensee Event Report LER 92-018 is hereby submitted in accordance with the requirements of 10CFR50.73. This event is of the type defined in the requirements per 10CFR50.73 (a) (2) (vi). This LER addresses a discovery made during the continued engineering assessment of cable tray separation. Additional cable configurations were identified that did not meet design criteria.

Very truly yours,

William A. Josiger
Resident Manager
Indian Point Three Nuclear Power Plant

WAJ/ED/EH
Attachment

cc: Mr. Thomas T. Martin
Regional Administrator
Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, GA 30339

310015 Senior Resident Inspector
Indian Point 3

IE22

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Potential compromise of redundant circuit separation caused by cable tray configuration not addressed in plant design

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 NAMES	DOCKET NUMBER(S)
1 1	1 7	9 2	9 2	0 1 8	0 0	1 2	2 4	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Edward Diamond, Senior Plant Engineer	9 1 4 7 3 6 - 8 0 4 5

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

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS
B	F A	C B L		N					
B	F A	T Y							

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single space typewritten lines) (16)

A walkdown of safety-related cables and cable trays conducted during the cycle 8/9 refueling outage, April to August 1992, identified cable configurations that did not meet design criteria. Fire barriers were installed where cable tray separation was inadequate. An analysis of the cables contained in these trays determined that a number of redundant, safety-related circuits were potentially compromised prior to the installation of the fire barriers. The root cause of the event was an original plant design process deficiency. Analysis and barrier installations were completed on November 18, 1992.

DESCRIPTION OF THE EVENT

A walkdown of safety-related cables and cable trays was conducted during the cycle 8/9 refueling outage, April to August 1992 as a continuing investigation of cable tray separation issues discovered in July of 1991, (Reference LER 50-286/91-008 and LER 50-286/91-008 Revision 1). The purpose of the plant walkdown was to ensure that cable separation criteria were satisfied, that either sufficient distance existed between cable trays or that a fire barrier was installed between the cable trays. The walkdown encompassed all plant areas.

The Indian Point 3 Final Safety Analysis Report section 8.4 states that separation of channels is established throughout the plant by the use of separate trays or conduits. In addition, whenever a heavy power tray is located less than three (3) feet beneath any tray of a different channel, a transite or marinite fire barrier is installed between the trays. A vertical barrier is installed where trays of different channels are installed less than one (1) foot apart, horizontally. Additionally, a horizontal barrier is installed where trays (other than heavy power) are installed less than one (1) foot beneath any tray of a different channel.

The walkdown also included identifying perpendicular trays, cables not in conduit, or cables not in trays that were in close proximity. If the walkdown identified inadequate cable tray separation, fire barrier installation was planned. Subsequently, an analysis of the cables contained in the trays with inadequate separation was conducted.

At 1700 hours on November 17, 1992, with the unit operating at full power, the engineering analysis of cable in trays with inadequate separation identified circuits which were found to be redundant, safety-related circuits potentially compromised by inadequate cable separation where perpendicular cable trays passed in proximity. This condition had existed since plant construction. Fire barriers had been installed for these circuits during the cycle 8/9 refueling outage, April through August, 1992. The NRC Operations Center was notified via the Emergency Notification System at 1740 hours. The NRC Resident inspector was also notified.

At 1533 on November 18, 1992, with the unit operating at full power, the engineering analysis of cables in trays with inadequate separation was completed.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The engineering analysis identified a perpendicular cable tray configuration containing redundant safety equipment cables that had a fire barrier planned for installation but not completed. The Technical Specification, safety-related equipment affected were the low pressure steam dump valves. These valves dump steam from the high pressure turbine exhaust headers to the condenser on a turbine trip. The purpose is to prevent low pressure turbine overspeed. The lack of a fire barrier for a short span where the redundant control circuit wiring cable trays passed in close proximity led to conservatively considering all six valves inoperable.

The low pressure steam dump valve inoperability necessitated a load reduction to less than 950 MWe which was started at 1545 on November 18, 1992. The load reduction was necessary to comply with Technical Specification 3.4.D. The NRC Operations Center was notified via the Emergency Notification System at 1632 hours. The NRC resident inspector was also notified. The load reduction was completed at 1630 hours with load stabilized at 945 MWe. Installation of a fire barrier was completed at 1728 hours. A load ascension to full power was begun at 1730, and full power was reached at 2030 hours.

Attachment I lists the redundant, safety-related circuits potentially compromised by inadequate cable separation.

Although all safety-related circuits have been verified to meet the design criteria for cable tray separation, installation of barriers continue for future plant modification considerations.

CAUSE OF THE EVENT

The cause of the cable separation barriers not installed between perpendicular cable trays passing in proximity was that cable tray separation for this configuration was not addressed in the original plant design. The root cause was that the design process was insufficient.

CORRECTIVE ACTIONS

A walkdown and engineering analysis to verify that cable separation criteria are satisfied has been performed for all plant areas. The analysis was completed on November 18, 1992. Marinite fire barriers were installed between the cable trays and cables.

ANALYSIS OF THE EVENT

This event is reportable under 10CFR50.73 (a) (2) (vi) as a discovery of construction and design inadequacies that could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

SAFETY SIGNIFICANCE

The cable tray separation design inadequacies described in this LER existed since plant construction. During that time the safety function of structures or systems that are needed to remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident were not challenged during a fire.

This event had no adverse impact on public health and safety.

Similar events were reported in LER 50-286-91-008 and LER 50-286-91-008, Revision 1.

SECURING FROM THE EVENT

A walkdown and engineering analysis to verify that cable separation criteria are satisfied has been performed for all plant areas. The engineering analysis and corrective actions were both completed on November 18, 1992

The load reduction following the inoperability of the low pressure condenser steam dump valves ended at 1730 hours on November 18, 1992. The unit returned to full power at 2030 on November 18, 1992.

ATTACHMENT I

Redundant, Safety-Related Circuits Potentially Compromised by Inadequate Cable Separation

Item Number	Equipment Description
1	Reactor Coolant Pump (RCP) Component Cooling Water (CCW) Supply Containment Isolation Valves
2	RCP Bearing Cooling CCW Return Containment Isolation Valves
3	RCP Thermal Bearing Cooling CCW Return Containment Isolation Valves
4	Residual Heat Removal (RHR) Miniflow Containment Isolation Valves
5	Primary Water Containment Isolation Valves
6	Condenser Air Ejector Effluent Containment Isolation Valves
7	Containment Airborne Radiation Monitor Inlet Containment Isolation Valves
8	Containment Airborne Radiation Monitor Outlet Containment Isolation Valves
9	Containment Airborne Radiation Monitor Inlet Containment Isolation Valves Weld Channel Supply Valves
10	Containment Airborne Radiation Monitor Outlet Containment Isolation Valves Weld Channel Supply Valves
11	Steam Generator (SG) 31 Blowdown Containment Isolation Valves

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

ATTACHMENT I

Redundant, Safety-Related Circuits Potentially Compromised by Inadequate Cable Separation

Item Number	Equipment Description
12	SG 31 Sample Containment Isolation Valves
13	Letdown Containment Isolation Valves
14	CCW Cooling to the Excess Letdown Heat exchanger Containment Isolation Valves
15	Pressurizer (Przr) Steam Space Sample Containment Isolation Valves
16	Przr Liquid Sample Containment Isolation Valves
17	Accumulator Sample Containment Isolation Valves
18	Reactor Coolant System Hot Leg Sample Containment Isolation Valves
19	Reactor Coolant Drain Tank (RCDT) Pumps Discharge Containment Isolation Valves
20	Containment Sump Pumps Discharge Containment Isolation Valves
21	RCDT Vent Header Containment Isolation Valves
22	Isolation Valve Seal Water Tank Discharge Valves (1410 and 1413)
23	SG 34 Blowdown Containment Isolation Valves
24	SG 33 Blowdown Containment Isolation Valves

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

ATTACHMENT I

Redundant, Safety-Related Circuits Potentially Compromised by Inadequate Cable Separation

Item Number	Equipment Description
25	RHR Throttling Valves
26	Low Head to High Head Recirculation Isolation Valves
27	Fan Cooler Unit (FCU) 31 Charcoal Filter Dousing Valves
28	FCU 32 Charcoal Filter Dousing Valves
29	FCU 33 Charcoal Filter Dousing Valves
30	FCU 35 Charcoal Filter Dousing Valves
31	RHR Heat Exchanger 31 Outlet Isolation Valves
32	RHR Heat Exchanger 32 Outlet Isolation Valves
33	Excess Letdown Stop Valves
34	Normal Letdown Stop Valves
35	31 and 33 Service Water Pumps
36	Reactor Coolant System (RCS) Loops 31 and 33 Hot Leg Resistance Temperature Detectors (RTDs)
37	SG 31, 32, 33, and 34 Low Feedwater Flow Inputs to the ATWS Mitigating System Actuation Circuitry (AMSAC)
38	Low pressure steam dump valves