Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan. New York 10511
914 736.8001

New York Power Authority

John H. Garrity
Resident Manager

December 17, 1993
IPN-93-160

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop PI-137
Washington, D.C. 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Licensee Event Report # 93-051 -00
"A Seismically Induced Failure Of A Fire Main, Caused By Personnel Error, Can Place The Plant Outside Design Basis"

Dear Sir:

The attached Licensee Event Report (LER) 93-051-00 is hereby submitted in accordance with the requirements of 10CFR50.73. This event is of the type defined in the requirements pursuant to 10CFR50.73(a)(2)(ii)(B). Also attached are the commitments made by the Authority in this LER.

Very Truly Yours,

JHGarrity

John H. Garrity
Resident Manager
Indian Point 3 Nuclear Power Plant

JHG/vjm

cc: See Next Page

IE22
1/1

Mr. Thomas T. Martin
Regional Administrator
Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

INPO Records Center
700 Galleria Parkway
Atlanta, Georgia 30339-5957

U.S. NRC Resident Inspector's Office
Indian Point 3

Attachment 1
List of Commitments

Number	Commitment	Due
IPN-93-160-01	Site Engineering will walkdown the fire protection system in category I areas to determine whether seismic class I components could be damaged by a seismically induced failure of the fire protection piping. Identified deficiencies will be corrected.	Startup
IPN-93-160-02	A modification will add seismic supports to the deluge system in the 480 volt switchgear room to prevent flooding due to an earthquake. The supports will be designed to maintain the pressure boundary during load conditions that include an earthquake.	Startup
IPN-93-160-03	An evaluation is being performed to determine the ability of the plant to maintain core cooling during the period required to restore power to a RHR pump following loss of the 480 volt switchgear due to an earthquake. The evaluation will establish the period available for returning power to the RHR pumps using the equipment powered from the 6.9 KV bus by the 10 CFR 50, Appendix R diesel generator. Conclusions regarding the effects on public health and safety can be made following this evaluation. LER 93-051-00 will be supplemented to discuss the evaluation and conclusions.	March 21, 1994

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point Unit 3	05000286	1 OF 6

TITLE (4)
A Seismically Induced Failure of A Fire Main, Caused By Personnel Error, Can Place The Plant Outside Design Basis

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
11	17	93	93	-- 051 --	00	12	17	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	000

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

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
John Whitney, Technical Services	(914) 736-8147

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
✓ YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		3	21	94

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

On November 17, 1993 at approximately 1845 hours with the plant in cold shutdown condition, Site Engineering identified a 10 inch diameter fire main in the Control Building that did not seem to be supported to seismic I requirements. This was evaluated through inspection and a one hour report was made to the NRC at 1921 hours. A seismically induced break of the fire main could result in flooding of the 480 volt switchgear. This is an unanalyzed condition and does not meet the IP3 FSAR requirement (i.e.,do not allow seismically induced failures in fire protection piping to damage seismic I components). The cause of this event is human error of indeterminate nature during the original design and installation. Compensating action was taken by isolating the header from outside the control building under administrative control. The fire protection system will be walked down in category I areas to determine extent of condition. Prior to startup, corrective actions will be taken to address any findings from the extent of condition walkdown and to add seismic I supports to prevent seismically induced rupture of the header in the 480 volt switchgear room. A supplement will be issued to fully identify safety significance.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Indian Point Unit 3	05000286	93	-- 051 --	00	2 OF 6

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

DESCRIPTION

On November 17, 1993 at approximately 1845 hours with the plant in cold shutdown condition (the reactor power level at 9 CPS, reactor coolant temperature at 106 degrees F, reactor coolant pressure at atmospheric and the pressurizer level at 26%), Site Engineering and the 480 volt switchgear System Engineer discussed a 10 inch diameter fire protection system (KP) fire main for the main transformer (XFMR) deluge system that did not appear to be seismically supported. DER 93-741 was written at 1845 hours to document this condition. Immediately afterwards, the System Engineer and Site Engineering inspected the fire main header, located in the 480 volt switchgear (SWGR) room at the 15 foot elevation of the control building (NA), and concluded that the 10 inch diameter fire main is supported to seismic class III criteria (i.e., non seismic). A seismic event could cause rupture. A preliminary calculation performed as part of the probabilistic risk assessment (PRA) indicates that a guillotine break would cause the 480 volt switchgear room to flood up to the critical height of 3 inches (at this level the 480 volt switchgear in the room would be lost) within one minute. No other safety related equipment would be affected by flooding from the event. Operations made a one hour report to the NRC on November 17, 1993 at 1921 hours.

Immediate corrective action was taken by closing the supply valve (valve FP-75) for the deluge system and putting the valve under administrative control (night order 93-342). The 10 inch deluge system is located in a small room within the 480 volt switchgear room. The valve is located outside the control building at the 15 foot elevation of the turbine building and can be opened by the Operations Department in the event of a fire at the transformers upon receipt of a fire alarm and after deenergizing the transformer.

The Final Safety Analysis Report (FSAR) criterion for the fire protection header does not allow a seismically induced failure of the fire protection piping to damage seismic category I components. An assessment of seismic capability performed by the Nuclear Engineering Department determined that the header could break in an earthquake. Technical Services evaluated the history of this event and traced the design and installation of the deluge system to the original plant construction. The FSAR criterion was not met at that time and this condition has therefore existed since initial plant operation. The circumstances surrounding the design error in construction could not be established. Inattention to detail is suspected because the deluge system does not protect any equipment located inside the control building and the fire main is enclosed in a small room with block walls and a doorway (no door). Since no damage to safety related equipment would occur if the fire main pipe were to fall and there was originally a door to the room, seismic restraints may have been overlooked.

A system interaction study performed in the early 1980s to study internal flooding induced by a seismic event (this study postulates a break in all seismic class III piping) also missed this deficient design. The fire main deluge system is out of sight behind the block wall enclosure and the failure to consider a piping break is suspected to be inattention to detail.

CAUSE OF THE EVENT

The cause of this event is personnel error of an indeterminate nature. Inattention to detail is suspected. A final determination was not considered necessary for corrective action.

CORRECTIVE ACTION

The procedures of the Authority's Modification Control Manual (MCM) program require the responsible engineering department to specify installation instructions. These instructions include incorporating seismic design criteria and walkdown of installations prior to system acceptance. The MCM program was not in place at the time of the event. Adherence to this MCM program assures that this event will not occur in the future.

The following corrective actions will be performed in order to establish extent of condition and address the deficiencies.

- Site Engineering will walkdown the fire protection system in category I areas to determine whether seismic class I components could be damaged by a seismically induced failure of the fire protection piping. Identified deficiencies will be corrected. This is scheduled for completion by startup.

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

- A modification will add seismic supports to the deluge system in the 480 volt switchgear room to prevent flooding due to an earthquake. The supports will be designed to maintain the pressure boundary during load conditions that include an earthquake. This modification is scheduled for completion by startup.

- An evaluation is being performed to determine the ability of the plant to maintain core cooling during the period required to restore power to a RHR pump following loss of the 480 volt switchgear due to an earthquake. The evaluation will establish the period available for returning power to the RHR pumps using the equipment powered from the 6.9 KV bus by the 10 CFR 50, Appendix R diesel generator. Conclusions regarding the effects on public health and safety can be made following this evaluation. LER 93-051-00 will be supplemented by March 21, 1994 to discuss the evaluation and conclusions.

ANALYSIS OF THE EVENT

This event is reportable under 10 CFR 50.73(a)(2)(ii)(B). The Licensee shall report any event or condition that resulted in the plant being in a condition outside the licensing design basis. A seismic event could rupture the fire main header in the 480 switchgear room and render the 480 volt safety related switchgear out of service due to flooding. This places the plant outside its design basis.

Similar events have been reported in previous Licensee Event Reports. Loss of emergency onsite AC power as a consequence of failures has been identified in LERs 93-048, 93-042, 93-027 and 93-026. The potential failure of the 480 volt switchgear due to support system failure is identified in LER 93-048. Reports identifying inadequacies in original design are LERs 93-048, 93-047, 93-045, 93-044, 93-035, 93-030, 93-026, 93-007, 92-008 and 92-006. The potential for seismically induced failures of plant safety systems was reported in LERs 93-036 and 93-027.

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

SAFETY SIGNIFICANCE

The effect of this event on the health and safety of the public is still under evaluation and will be reported in a supplement.

With the deluge fire system not seismically supported, the 480 volt safety related switchgear could be lost as a consequence of an earthquake. Loss of the 480 volt switchgear would render the emergency diesel generators (EDG) ineffective (the EDGs would start but there would be no power distribution).

Following the earthquake, there are three potential sources of power:

- The expected availability of the 10 CFR 50, Appendix R diesel generator is discussed in LER 93-027. This diesel generator is expected to survive an earthquake because of the low intensity of the IP3 earthquake (i.e., 0.15g horizontal and 0.1 g vertical) and the inherent capabilities identified through a review of historical data during the resolution of Unresolved Safety Issue A-46. There are control room procedures in place to allow shutdown using the 10 CFR 50, Appendix R diesel generator to feed the 6.9 KV bus. The 6.9 KV bus would not be affected by the postulated flooding of the 480 volt switchgear room.

- There are also control room procedures in place to allow the Consolidated Edison gas turbines to be connected to the 13.8 KV bus, this bus would then feed the 6.9 KV bus. One or more of these gas turbines (there are three) would be expected to survive the earthquake based on an inherent capability similar to that of the 10 CFR 50, Appendix R diesel generator. The gas turbines will provide sufficient power to use non safety components, if any are available.

- The remaining power source would be offsite power. The offsite power grid has some inherent capability to survive an earthquake. The uniform building code is applicable to design and requires consideration of earthquakes and high winds. Availability is difficult to assess due to the potential for grid instabilities and damage from failures outside grid boundaries. If offsite power is not lost during the postulated seismic event there are control room procedures in place to align this power source to the 6.9 KV bus and sufficient power will be available to use non safety components, if any are available.

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Indian Point Unit 3	05000286	93	-- 051 --	00	6 OF 6

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

The 6.9 KV bus will provide power for safe shutdown in accordance with 10 CFR 50, Appendix R. However, the earthquake induced failure of the 480 volt switchgear would require substantial time to repair. Initiation of the residual heat removal (RHR) system may be necessary to cope with the event due to this repair time.

Power to the RHR pumps is lost when the 480 volt switchgear is lost. The RHR valves can be manually aligned, if required. Power to the pumps can be restored if either the 312 or 313 non safety grade 480 volt switchgear survive the earthquake. Survival is expected because of the low intensity of the IP3 earthquake (i.e., 0.15g horizontal and 0.1 g vertical) and the inherent capability of switchgear to survive a seismic event (historical data during the resolution of Unresolved Safety Issue A-46 indicate no more than relay chatter is expected). The 6.9 KV bus provides power to the 312 and 313 switchgear which are located in the turbine building at the 15 foot elevation. Cable, currently available onsite, could be routed from either non safety 480 volt switchgear to a RHR pump feeder (approximately 100 feet with no major obstacle intervening) using existing procedures for cable routing and repair. Maintenance estimated this work could be completed in approximately 15 hours.

Reactor Engineering is currently evaluating the ability of the plant to maintain core cooling during the period required to restore RHR. Station blackout evaluations performed by the industry indicate that natural circulation can be used to cool the core while water is available to the primary and secondary system. The evaluation will establish the period available for repair using the equipment powered from the 6.9 KV bus by the 10 CFR 50, Appendix R diesel generator. This LER will be supplemented to discuss the evaluation and present conclusions regarding the effects on public health and safety.

The extent of condition for this event is being determined by an engineering walkdown of the fire protection systems to assure that no other seismic class I components could be affected by a seismically induced failure. No additional review is considered necessary because of the seismic interaction study has already assessed the potential for seismically induced failures and the ongoing IPE is looking at the probability of pipe rupture and the effects.