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L. M. Hill
Resident Manager

January 9, 1995
IPN-95-001

U.S. Nuclear Regulatory Commission
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SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Licensee Event Report # 93-051-01
"A Seismically Induced Failure Of A Fire Main, Caused By Personnel Error, Can Place The Plant Outside Design Basis"

Dear Sir:

The attached supplemental Licensee Event Report (LER) 93-051-01 is submitted as required by 10CFR50.73. This event is of the type defined in 10CFR50.73(a)(2)(ii)(B). The supplement provides the status of commitments, editorially corrects the LER and completes the evaluation of safety significance. There are no new commitments made by the Authority in this LER.

Very truly yours,

L. M. Hill
Resident Manager
Indian Point 3 Nuclear Power Plant

LMH/vjw

Attachments
cc: See next page

IE22
1/1

cc: Mr. Thomas T. Martin
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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

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

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
11	17	93	93	-- 051 --	01	01	09	95		05000
									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)
Stephen Prussman, Senior Nuclear Licensing Engineer	(914) 736-8856

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				

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 have resulted in flooding of the 480 volt switchgear. This was an unanalyzed condition and did not meet the Indian Point 3 (IP3) Final Safety Analysis Report (FSAR) requirement (i.e., do not allow seismically induced failures in fire protection piping to damage seismic class I components). The cause of this event was 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. To determine extent of condition, the fire protection system was walked down in category I areas. Corrective action was taken by adding seismic class I supports to prevent seismically induced rupture of the header in the 480 volt switchgear room. The effect on the public health and safety was not significant.

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 was supported to seismic class III criteria (i.e., non seismic). A seismic event could have caused rupture. A preliminary calculation performed as part of the probabilistic risk assessment (PRA) indicates that a guillotine break would have caused 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 have been 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 class I components. An assessment of seismic capability performed by the Nuclear Engineering Department determined that the header could have broken during a design basis 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 have occurred if the fire main pipe fell 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 have been performed in order to establish extent of condition and address the deficiencies.

- The Nuclear Engineering Department has completed a walkdown of the fire protection system in category I areas and determined that piping is seismically supported so that there are no operability concerns (some discrepancies require correction). This completes the action in commitment IPN-93-160-01.

- The deluge system in the 480 volt switchgear room has been seismically supported to maintain the pressure boundary during load conditions that include an earthquake. This completes the action in commitment IPN-93-160-02.

- The evaluation of safety significance was completed and included in this LER supplement (LER 93-051-01). This completes the action in commitment IPN-93-160-03.

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 have ruptured the fire main header in the 480 volt switchgear room and rendered the 480 volt safety related switchgear out of service due to flooding. This placed 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.

SAFETY SIGNIFICANCE

The event did not significantly affect the public health and safety.

With the deluge fire system not seismically supported, the 480 volt safety related switchgear could have been lost as a consequence of an earthquake. Loss of the 480 volt switchgear would have rendered 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 Indian Point 3 (IP3) earthquake (i.e., 0.15g horizontal and 0.1g 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 have been 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 and 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 was 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 would have been available to use non safety components, if any were available.

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 have required substantial time to repair. Initiation of the residual heat removal (RHR) system may have been necessary to cope with the event due to this repair time.

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 --	01	6 OF 6

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

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 and 312A 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 0.1g 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, available onsite, could have been 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.

Time would have been available to repair the cabling to the RHR pump. There are adequate water supplies and equipment is available to maintain a stable core configuration whether operating or shutdown. There are certain shutdown situations (the first two days after shutdown when in mid loop operation or with all the steam generators drained) where normal water supplies are not adequate but the refueling water storage tank (TK) is available for core makeup by gravity drain through the RHR system. The plant was shutdown 7.36 years during the 17.85 years from initial criticality to addition of the seismic restraint on fire header. The average annual probability of a design basis earthquake (DBE) while shutdown during that period is 8.65E-5. The probability of the DBE during the first several days is much less and is not considered reasonable and credible.

The plant was shutdown 4.2 years during the 10.35 years from initial criticality to the addition of the Appendix R diesel. The average annual probability of a design basis earthquake while shutdown during that period is 8.5E-5. The event was not considered reasonable and credible.

The engineering walkdown of the fire protection systems assured that there are no other seismic class I components that could be affected. No additional review is considered necessary because the seismic interaction study has already assessed the potential for seismically induced failures and the ongoing Individual Plant Evaluation is looking at the probability of pipe rupture and the effects.