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

NewYorkPower Authority

Robert J. Barrett
Site Executive Officer

August 28, 1998
IPN-98-093

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

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Request for Additional Information Regarding Summary Report on the Verification of Seismic Adequacy of Mechanical and Electrical Equipment (TAC No. M69454)

REFERENCE: NRC Letter, G. F. Wunder to J. Knubel, dated March 9, 1998, "Request for Additional Information Regarding Summary Report on the Verification of Seismic Adequacy of Mechanical and Electrical Equipment – Indian Point Nuclear Generating Unit No. 3."

A025

Dear Sir:

Attached is the information requested by the referenced correspondence.

There are no commitments made by the Authority in this letter. If you have any questions, please contact Mr. Ken Peters at (914) 736-8029.

Very truly yours,

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

cc: See next page

9809090299 980828
PDR ADOCK 05000286
P PDR

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King Of Prussia, Pennsylvania 19406-1415

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

Attachment I to IPN-98-093

Response to Request for Additional Information

Regarding Summary Report on the Verification of

Seismic Adequacy of Mechanical and Electrical Equipment

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
DPR-64

Question a

Describe what reviews were performed to determine if any local operator actions required to safely shutdown the reactor could be affected by potentially adverse environmental conditions (such as loss of lighting, excessive heat or humidity, or in-plant barriers) resulting from the seismic event. Describe how staffing was evaluated and describe the reviews which were conducted to ensure operators had adequate time and resources to respond to such events.

Response a

As described in Generic Implementation Procedure, Revision 2 (GIP-2) (reference 1), Part II, Section 3.2.5, the only potential events which must be considered in the Unresolved Safety Issue (USI) A-46 program are a safe shutdown earthquake (SSE) and loss of offsite power (LOOP). The plant operating procedures used to shut down the reactor following a LOOP have previously been validated for local operator actions as one of the FSAR Chapter 14 accident scenarios.

The potential for failure of plant structures and equipment is not considered credible at eastern U.S. earthquake levels. Earthquake experience has shown that typical industrial structures are able to withstand earthquakes larger than the SSEs for Eastern U.S. nuclear plants without collapse or failure. The potential for local failure of architectural features (such as suspended ceilings in the control room) and the potential for adverse seismic spatial interactions in the vicinity of safe shutdown equipment, where local operator actions may be required, was explicitly evaluated as required in GIP-2, Part II, Section 4.5 and Appendix D. For example, this review included a check that the masonry walls near safe shutdown equipment are seismically adequate based on the results of the NRC Bulletin 80-11 program.

The systems and equipment selected for seismic review in the USI A-46 program are those for which Normal, Off-Normal, and Emergency Operating Procedures are available to bring the plant from a normal operating mode to a hot shutdown condition. As required by GIP-2, Part II, Sections 3.2.8 and 3.7, the safe shutdown equipment list (SSEL) was reviewed by the plant Operations Department to confirm that it is compatible with these plant procedures. Since these plant procedures had already been validated to ensure that adequate time and resources are available for operators to respond to a LOOP incident, it was not necessary to re-validate these procedures for the USI A-46 program.

The only additional operator actions, beyond those associated with the LOOP accident scenario, which must be performed to bring the plant from a normal operating mode to a hot shutdown condition are those that could result from the vibratory motion of the SSE as identified in our submittal.

The operator actions that might have to be performed are in an area that is continuously manned, the Control Room, or in an area that the operator would have to be in to respond to an accident that has already been evaluated. Additionally, indication would be available to the operator that would provide him information on what equipment had changed state.

Question b

As part of your review, were any control room structures which could impact the operators ability to respond to the seismic event identified? Such items might include control room ceiling tiles, non-bolted cabinets, and non-restrained pieces of equipment (e.g., computer keyboards, monitors, stands, printers,). Describe how each of these potential sources of interactions has been evaluated and describe the schedule for implementation of the final resolutions.

Response b

The equipment located in the Control Room and evaluated as part of the Indian Point 3 A-46 review is listed in Table 1 (Attachment II). This table was obtained from Appendix D of reference 2, which was submitted to the NRC as an enclosure to reference 3. This table also shows the SQUG outlier issues for equipment that did not meet the SQUG GIP criteria. The outlier issues were also obtained from Table 1 "Outlier Summary" of reference 2.

The equipment listed in Table 1 was evaluated for interaction concerns. The method used for evaluating these potential sources of seismic spatial interaction is described in GIP-2, Part II, Section 4.5 and Appendix D. After performing this review, the Authority concluded that the control room equipment and the structures that are considered to be in the interaction area passed the GIP screening criteria except for the outliers that are shown in this table. Most of the outliers shown in Table 1 have already been resolved. As stated in this table and the letter of reference 3, the remaining outliers will be resolved prior to start-up from refueling outage 10.

The Authority believes that none of the remaining outliers listed in Table 1 would adversely impact the operator's ability to respond to an SSE.

The Authority did not consider the Control Room ceiling tiles as a source of potential interaction, because the Control Room eggcrate ceiling supports had been improved by a modification (83-3-201).

Question c

Describe what reviews were performed to determine if any local operator actions were required to reposition "bad actor relays." For any such activities describe how adverse environmental conditions (such as loss of lighting, excessive heat or humidity, or in-plant barriers) resulting from the seismic event were analyzed. Describe how staffing was evaluated and describe the reviews which were conducted to ensure operators had adequate time and resources to respond to such events.

Response c

Indian Point 3 does not have any "bad actor" relays in any control or power supply circuits associated with SSEL listed equipment. This is stated in Section 3.6 of reference 4, which was submitted to the NRC as enclosure 1 of reference 3. The conclusions reached in this referenced section are repeated below:

"The review consisted of a review of those documents identified as associated with the equipment listed on the SSEL, and other documents which provided listings of the types of relays used throughout the plant. NYPA personnel, both in engineering and at the IP-3 site were consulted in order to further substantiate the findings.

The results of the review determined that no essential relays within the control supply circuits associated with SSEL equipment are of the low rugged type listed in Appendix E of reference 5.

The review also determined that neither "Bad Actor" relays listed in Appendix E of reference 5, English Electric YCG or GE IJD (non 1E), were used in any control or power supply circuits associated with SSEL listed equipment."

Therefore this question does not apply to Indian Point 3 Nuclear Power Plant.

Question d

Describe which of the operator actions associated with resetting safe shutdown equipment list (SSEL) equipment affected by postulated relay chatter are considered to be routine and consistent with the skill of the craft. If not considered skill of the craft, what training and operational aids were developed to ensure the operators will perform the actions required to reset affected equipment?

Response d

The Authority's response (reference 3) identifies the additional actions that the operators might have to take to respond to a SSE. The actions that might have to be taken are actions that are routine and considered skill of the craft for an operator or are already addressed in normal, off-normal, or emergency operating procedures.

Question e

Assume the alarms associated with "bad actor relays" are expected to annunciate during the seismic event. Do the operators have to respond to those annunciators and review the annunciator response procedures associated with them for potential action? How would those additional actions impact the operators ability to implement the normal, abnormal, and emergency operating procedures required to place the reactor in a safe shutdown condition?

Response e

Indian Point 3 does not have any "bad actor" relays in any control or power supply circuits associated with SSEL listed equipment (see Response c).

Regarding spurious earthquake induced alarms in general, the earthquake motion is assumed to last less than a minute and the causes of the spurious alarms to have gone away while the operators are responding to a turbine trip and reactor scram. The NRC staff and SQUG representatives discussed this topic in detail, including discussions held at a meeting on August 3, 1988. The results of that evaluation and review are summarized in EPRI NP-7148, Section 3.5.3, where the following conclusion is reached: "Accordingly, there appear to be no reasonable bases or evidence which would suggest that spurious alarms resulting from an earthquake may lead to abnormal operator responses. Therefore, special operating procedures or relay evaluation actions to address potential spurious alarms are not considered warranted and relays affecting alarms need not be seismically adequate."

The NRC staff accepted the relay functionality review procedures summarized in GIP-2 and described in detail in EPRI NP-7148 (including the above conclusion) in Supplemental Safety Evaluation Report No. 2 on GIP-2. Therefore, the Authority does not consider it necessary to perform additional reviews of the effect spurious alarms caused by relays or other causes as a result of a seismic event.

Question f

To the extent that normal, abnormal, and emergency operating procedures were modified to provide plant staff with additional guidance on mitigating the A-46 Seismic Event, describe what training was required and provided to the licensed operators, non-licensed operators, and other plant staff required to respond to such events.

Response f

The Authority has not revised any normal, off-normal, or emergency operating procedures as a result of USI A-46.

REFERENCES

1. SQUG Generic Implementation Procedure, Revision 2 (GIP-2), dated February 14, 1992.

2. Atometrics Report No. 18904-01, "Seismic Evaluation Report Verification of the Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors in Response to NRC Generic Letter 87-02/Unresolved Safety Issue (USI) A-46 for the Indian Point Nuclear Generating Station Unit No. 3", dated October 23, 1995

3. NYPA Letter IPN-95-118 From W.J. Cahill, Jr. to U.S. Nuclear Regulatory Commission, "Indian Point 3 Nuclear Power Plant Docket No. 50-286 Summary Reports for the Resolution of Unresolved Safety Issue A-46 Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors", dated November 16, 1995.

4. Atometrics Report No. 18904-00, "Safe Shutdown Equipment Selection and Relay/Contact Evaluations for the Seismic Verification of Equipment in Response to Unresolved Safety Issue (USI) A-46 for the Indian Point Nuclear Generating Station Unit No. 3", dated October10, 1995.

5. EPRI Report Number NP-7148-SL, "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality", Final Report, December 1990.

Attachment II to IPN-98-093

Table I

SQUG Reviewed Equipment Located in the Control Room

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
DPR-64

TABLE I
SQUG REVIEWED EQUIPMENT LOCATED IN THE CONTROL ROOM

ITEM	EQ. CL.	EQUIPMENT ID NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	INTERACT OK?	SQUG OUTLIER	OUTLIER ISSUE	RESOLUTION DATE
1	0	T1	CR AC THERMOSTAT	CB	53'-0"	NO	YES	Thermostat is located near a potential hazard (map).	Already resolved
2	14	K50	118 VAC INST BUS 31A	CB	53'-0"	YES	NO		
3	14	K51	118 VAC INST BUS 32A	CB	53'-0"	YES	NO		
4	14	PE6	118 VAC INSTR BUS 34	CB	53'-0"	YES	NO		
5	14	PE7	118 VAC INSTR BUS 33	CB	53'-0"	YES	NO		
6	14	PE8	118 VAC INST BUS 31	CB	53'-0"	YES	NO		
7	14	PE9	118 VAC INST BUS 32	CB	53'-0"	YES	NO		
8	14	PNL K48	125 VDC DISTRIBUTION PNL 31A	CB	53'-0"	YES	NO		
9	14	PNL K49	125 VDC DISTRIBUTION PNL 32A	CB	53'-0"	YES	NO		
10	14	PNL PC3	125 VDC DISTRIBUTION PNL 31	CB	53'-0"	YES	NO		
11	14	PNL PC4	125 VDC DISTRIBUTION PNL 32	CB	53'-0"	YES	NO		
12	14	PNL PC8	125 VDC DISTRIBUTION PNL 33	CB	53'-0"	YES	NO		
13	14	PNL PD9	125 VDC DISTRIBUTION PNL 34	CB	53'-0"	YES	NO		
14	20	CAB J01	CTMT MONITORING CABINET CHANNEL I	CB	53'-0"	YES	NO		
15	20	CAB J02	CTMT MONITORING CABINET CHANNEL II	CB	53'-0"	YES	NO		
16	20	CAB JR9	RVLIS PANEL	CB	53'-0"	YES	NO		
17	20	FLIGHT PANEL	FLIGHT PANEL	CB	53'-0"	NO	YES	1. Loose relays in Panel Section FBF 2. Demand metering panel is not anchored 3. Adjacent panels J01 and J02 are not connected to flight panel, nor to each other	1. Already resolved 2. Already resolved 3. Will be resolved prior to start up from Refueling Outage 10.
18	20	RACK A-1	CCR RACK	CB	53'-0"	YES	NO		
19	20	RACK A-10	CCR RACK	CB	53'-0"	YES	NO		
20	20	RACK A-2	CCR RACK	CB	53'-0"	YES	NO		
21	20	RACK A-3	CCR RACK	CB	53'-0"	YES	NO		
22	20	RACK A-4	CCR RACK	CB	53'-0"	YES	NO		
23	20	RACK A-5	CCR RACK	CB	53'-0"	YES	NO		
24	20	RACK A-6	CCR RACK	CB	53'-0"	YES	NO		
25	20	RACK A-7	CCR RACK	CB	53'-0"	YES	NO		
26	20	RACK A-8	CCR RACK	CB	53'-0"	YES	NO		
27	20	RACK A-9	CCR RACK	CB	53'-0"	YES	NO		
28	20	RACK B-1	CCR RACK	CB	53'-0"	YES	NO		
29	20	RACK B-10	CCR RACK	CB	53'-0"	YES	NO		
30	20	RACK B-11	CCR RACK	CB	53'-0"	YES	NO		
31	20	RACK B-2	CCR RACK	CB	53'-0"	YES	NO		
32	20	RACK B-3	CCR RACK	CB	53'-0"	YES	NO		
33	20	RACK B-4	CCR RACK	CB	53'-0"	YES	NO		

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SQUG REVIEWED EQUIPMENT LOCATED IN THE CONTROL ROOM

ITEM	EQ. CL.	EQUIPMENT ID NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	INTERACT OK?	SQUG OUTLIER	OUTLIER ISSUE	RESOLUTION DATE
34	20	RACK B-5	CCR RACK	CB	53'-0"	YES	NO		
35	20	RACK B-6	CCR RACK	CB	53'-0"	YES	NO		
36	20	RACK B-7	CCR RACK	CB	53'-0"	YES	NO		
37	20	RACK B-8	CCR RACK	CB	53'-0"	YES	NO		
38	20	RACK B-9	CCR RACK	CB	53'-0"	YES	NO		
39	20	RACK C-1	CCR RACK	CB	53'-0"	YES	NO		
40	20	RACK C-10	CCR RACK	CB	53'-0"	YES	NO		
41	20	RACK C-11	CCR RACK	CB	53'-0"	YES	YES	Rack not anchored. Base channel not bolted.	Already resolved.
42	20	RACK C-2	CCR RACK	CB	53'-0"	YES	NO		
43	20	RACK C-3	CCR RACK	CB	53'-0"	YES	NO		
44	20	RACK C-4	CCR RACK	CB	53'-0"	YES	NO		
45	20	RACK C-5	CCR RACK	CB	53'-0"	YES	NO		
46	20	RACK C-6	CCR RACK	CB	53'-0"	YES	NO		
47	20	RACK C-7	CCR RACK	CB	53'-0"	YES	NO		
48	20	RACK C-8	CCR RACK	CB	53'-0"	YES	NO		
49	20	RACK C-9	CCR RACK	CB	53'-0"	YES	NO		
50	20	RACK D-1	CCR RACK	CB	53'-0"	YES	NO		
51	20	RACK D-10	CCR RACK	CB	53'-0"	YES	NO		
52	20	RACK D-11	CCR RACK	CB	53'-0"	YES	NO		
53	20	RACK D-2	CCR RACK	CB	53'-0"	YES	NO		
54	20	RACK D-3	CCR RACK	CB	53'-0"	YES	NO		
55	20	RACK D-4	CCR RACK	CB	53'-0"	YES	NO		
56	20	RACK D-5	CCR RACK	CB	53'-0"	YES	NO		
57	20	RACK D-6	CCR RACK	CB	53'-0"	YES	NO		
58	20	RACK D-7	CCR RACK	CB	53'-0"	YES	NO		
59	20	RACK D-8	CCR RACK	CB	53'-0"	YES	NO		
60	20	RACK D-9	CCR RACK	CB	53'-0"	YES	NO		
61	20	RACK E-1	CCR RACK	CB	53'-0"	YES	NO		
62	20	RACK E-2	CCR RACK	CB	53'-0"	YES	NO		
63	20	RACK E-3	CCR RACK	CB	53'-0"	YES	NO		
64	20	RACK E-4	CCR RACK	CB	53'-0"	YES	NO		
65	20	RACK E-5	CCR RACK	CB	53'-0"	YES	NO		
66	20	RACK E-6	CCR RACK	CB	53'-0"	YES	NO		
67	20	RACK E-7	CCR RACK	CB	53'-0"	YES	NO		
68	20	RACK F-1	CCR RACK	CB	53'-0"	YES	NO		
69	20	RACK F-2	CCR RACK	CB	53'-0"	YES	NO		

TABLE I
SQUG REVIEWED EQUIPMENT LOCATED IN THE CONTROL ROOM

ITEM	EQ. CL.	EQUIPMENT ID NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	INTERACT OK?	SQUG OUTLIER	OUTLIER ISSUE	RESOLUTION DATE
70	20	RACK F-3	CCR RACK	CB	53'-0"	YES	NO		
71	20	RACK F-4	CCR RACK	CB	53'-0"	YES	NO		
72	20	RACK F-5	CCR RACK	CB	53'-0"	YES	NO		
73	20	RACK F-6	CCR RACK	CB	53'-0"	YES	NO		
74	20	RACK F-7	CCR RACK	CB	53'-0"	NO	YES	Adjacent cabinet Rack F-6 not attached to Rack F-7 and F-7 has essential relays mounted in it.	Will be resolved prior to start up from Refueling Outage 10.
75	20	RACK G-1	CCR RACK	CB	53'-0"	YES	NO		
76	20	RACK G-2	CCR RACK	CB	53'-0"	YES	YES	Relay LC-112C/X loose.	Already resolved.
77	20	RACK G-3	CCR RACK	CB	53'-0"	YES	YES	ST1, ST11, and 2-SI-DI common mounting plate loose.	Already resolved.
78	20	RACK G-4	CCR RACK	CB	53'-0"	YES	NO		
79	20	RACK G-5	CCR RACK	CB	53'-0"	YES	NO		
80	20	RACK G-6	CCR RACK	CB	53'-0"	YES	NO		
81	20	RACK H-1	CCR RACK	CB	53'-0"	YES	NO		
82	20	RACK H-2	CCR RACK	CB	53'-0"	YES	YES	Bolt tightness check revealed loose nuts.	Already resolved.
83	20	RACK H-3	CCR RACK	CB	53'-0"	YES	YES	Bolt tightness check revealed loose nuts.	Already resolved.
84	20	RACK H-4	CCR RACK	CB	53'-0"	YES	NO		
85	20	RACK H-5	CCR RACK	CB	53'-0"	YES	NO		
86	20	SUPERVISORY PANEL	SUPERVISORY PANEL	CB	53'-0"	YES	YES	Panel not anchored on the back side.	Already resolved.