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New York Power Authority

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Nuclear Generation

September 21, 1993
IPN-93-109

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

Attn: Document Control Desk

Subject: **Indian Point 3 Nuclear Power Plant**
Docket No. 50-286
Response to Generic Letter 93-04
Rod Control System Failure and
Withdrawal of Rod Control Cluster Assemblies

References:

1. NRC Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54 (f)", dated June 21, 1993.

2. Letter from Ashok C. Thadani (Director, Division of Systems and Safety and Analysis, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission) to Roger Newton (Chairman, Westinghouse Owners Group, Regulatory Response Group), dated July 26, 1993, "WOG Request for Schedular Relief in Responding to NRC Generic Letter 93-04".

3. WCAP-13803, Revision 1, "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal".

Dear Sir:

This letter provides the Authority's 90 day response to Generic Letter (GL) 93-04 (Reference 1) for the Indian Point 3 Nuclear Power Plant. This generic letter was addressed to all licensees with the Westinghouse Rod Control System for action.

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As a result of this generic letter and subsequent correspondence between the Westinghouse Owner's Group (WOG) and the NRC, licensees were required to submit two letters to the NRC addressing the potential occurrence of an asymmetric rod withdrawal event at their facility. The first letter, to be submitted within 45 days of the date of the generic letter, was to outline short term actions that would be implemented to preclude any adverse affect on safety resulting from a postulated asymmetric rod withdrawal. The second letter, to be submitted within 90 days of the date of the generic letter, was to address: (1) whether or not the licensing basis for each facility was satisfied with regard to the requirements for rod control system response to a single failure, as stipulated in the General Design Criteria (GDC 25), and (2) any long term actions.

The Authority's 45 day response to Generic Letter 93-04, dated August 5, 1993, stated the Authority would confirm the applicability of WCAP-13803, entitled "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal". The Authority has completed this review, and concludes that the results of this effort are applicable to Indian Point 3; acceptable fuel design limits will not be exceeded. As such, the safety significance of such an asymmetric rod withdrawal event is minimal.

Attachment I of this letter provides an assessment of Indian Point 3's compliance with its licensing basis as it pertains to the rod control system's response to a single failure. Even though the analysis documented in WCAP-13803 indicates that Indian Point 3 is in compliance with its licensing basis, and that an asymmetric rod withdrawal event would result in minimal safety significance, the Authority believes that reasonable measures should be taken to minimize the potential for rod control system malfunctions. By taking such measures, the safety margin conservatively incorporated into the core design will be preserved and unnecessary challenges to safety will be reduced.

Attachment II provides a discussion of enhancements that will be made at Indian Point 3 to reduce the probability of an asymmetric rod withdrawal event. Attachment III provides a list of the commitments being made by the Authority with this submittal. For convenient reference, Attachment IV provides a list of the commitments that the Authority made in its August 5, 1993 submittal. These actions will increase the Indian Point 3 technical staff's awareness of the potential for an asymmetric rod withdrawal, as well as provide appropriate guidance in the event such a situation should arise. The Authority believes that the combined effect of the actions listed in Attachment III and IV will ensure the continued safe operation of Indian Point 3.

The Authority will implement both the short term actions listed in Attachment IV to this submittal, as well as the surveillances listed in Attachment III, prior to restart. As previously committed to in the Authority's August 5, 1993 submittal, as new information regarding this issue becomes available, the Authority will review its applicability to Indian Point 3, as well as re-evaluate the appropriateness of the actions committed to.

If you have any questions regarding this matter, please contact Mr. P. Kokolakis.

Very truly yours,

Ralph E. Beedle

STATE OF NEW YORK
COUNTY OF WESTCHESTER
Subscribed and Sworn to before me
this 21st day of September 1993

Kathleen D. Gallagher
Notary Public

Attachment

cc: See next page

KATHLEEN D. GALLAGHER
Notary Public, State of New York
No. 5004481
Qualified in Westchester County
Commission Expires Nov. 16, 1994

att: as stated

cc: U.S. Nuclear Regulatory Commission
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Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
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ATTACHMENT I TO IPN-93-109

ASSESSMENT OF INDIAN POINT 3's

COMPLIANCE WITH ITS LICENSING BASIS AS IT PERTAINS TO

SINGLE FAILURE VULNERABILITY OF WESTINGHOUSE ROD CONTROL SYSTEM

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
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Indian Point 3's design was based on the Atomic Industrial Forum's (AIF) interpretations and clarifications of the proposed General Design Criteria (GDC) published in the July 11, 1967 Federal Register. However, the NRC concluded in the IP3 Safety Evaluation Report (SER) that the plant design conformed to the intent of the GDC published in July 1971. While some differences exist between the proposed and final versions of the GDC, Criterion 25, "Protection System Requirements for Reactivity Control Malfunction" reads identically in both versions of the GDC. Specifically, Criterion 25 states:

> The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Consistent with the NRC's interpretation of GDC 25, the Authority has historically interpreted the fuel design limit referenced in GDC 25 to be the Departure from Nucleate Boiling (DNB) design basis. This interpretation is documented in the Authority's study of Indian Point 3's compliance with current safety rules and regulations, submitted to the NRC on August 11, 1980. However, until the Salem asymmetric rod withdrawal event, the occurrence of an asymmetric rod withdrawal event, or the possibility that such an event could be created by a single malfunction, was never considered to be a potentially credible event at Indian Point 3. As such, past analyses have not addressed this scenario for Indian Point 3.

In light of the recent asymmetric rod withdrawal event at Salem, the Authority reviewed the results of the Westinghouse/WOG efforts in this regard as they pertain to Indian Point 3. This review focused on two areas; the first area being the site specific DNB margins generated by the analysis documented in WCAP-13803, Revision 1, entitled "Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal", the second being the results from the testing performed by the WOG at the Salem training facility.

Westinghouse's safety analysis of a worst case asymmetric rod withdrawal event, as documented in WCAP-13803, Revision 1, demonstrated that DNB does not occur for any Westinghouse plant. Based upon the Authority's review of WCAP-13803, Revision 1, and the site specific data used to generate it, the Authority concludes that Indian Point 3 is bounded by Westinghouse's analysis. As such, DNB does not occur for a worst-case asymmetric rod withdrawal. However, it should be noted that the methodology utilized in this analysis has been submitted to the NRC by Westinghouse, but not formally approved by the NRC.

Based upon the Authority's review of the WOG test program, it is the Authority's understanding that a single malfunction will not result in a Salem type event. Rather, a single malfunction must be coupled with the existence of secondary phenomena, such as environmental factors (e.g., increased temperatures in the power supply cabinets) and differences in equipment response times (i.e.; response times are not precise, but have an acceptable band). As such, a single malfunction of the reactivity control system will not initiate a rod withdrawal event.

In conclusion, the Power Authority has reviewed the results of the Westinghouse generic safety analysis and the WOG's test program as they apply to Indian Point 3. The Westinghouse generic safety analysis concludes that Indian Point 3 is bounded by the results of these efforts. Acceptable fuel design limits (DNB) will not be exceeded during an asymmetric rod withdrawal event. In addition, the occurrence of such an event being initiated solely by a single malfunction is not credible. Based on this information, the Authority believes Indian Point 3 is in compliance with the requirements of GDC 25.

ATTACHMENT II TO IPN-93-109

ENHANCEMENTS THAT WILL BE MADE TO PRECLUDE THE OCCURRENCE OF AN ASYMMETRIC ROD WITHDRAWAL EVENT AT INDIAN POINT 3

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As stated in Attachment I, the Power Authority believes Indian Point 3 is in compliance with the requirements of GDC 25. The Power Authority also believes that measures should be taken to minimize the possibility of an asymmetric rod withdrawal event from occurring. Such events could erode the safety margin incorporated in the design of the core, as well as inherently challenge plant safety by challenging the operators and plant systems.

For this reason, prior to restart from the current outage, the following surveillance will be performed. For both insertion and withdrawal, the lift, movable and stationary currents for each rod will be monitored in the power cabinets. Not only will this verify that correct current orders are being generated prior to restart, but will provide up to date baseline readings for future comparisons.

For future restarts, if extensive maintenance and/or modification of the Control Rod Drive Control System is performed, this surveillance will be repeated for all rods affected by the maintenance. Routine maintenance, such as the replacement of an alarm card, which is not believed to contribute to the occurrence of an asymmetric rod withdrawal event, would not result in the performance of this surveillance. In addition, since power supply malfunctions have been identified as a potential contributing factor to the occurrence of the Salem asymmetric rod withdrawal event, power supply repairs/replacement would require performing this surveillance for all rods affected.

Westinghouse is currently developing details of a modification for the Rod Control System current order timing to prevent an uncontrolled asymmetric rod withdrawal event. As information regarding this proposed modification becomes available, the Authority will evaluate the appropriateness of this modification for Indian Point 3. Once the Authority has decided upon a long term course of action (system modification, site specific analysis, etc.) based upon our evaluation, the NRC will be advised.

The Authority believes that these measures, combined with the commitments made in the Authority's August 5, 1993 submittal, (see Attachment IV) will adequately ensure that an asymmetric rod withdrawal event will not occur at Indian Point 3. The commitments made by the Authority in the August 5, 1993 submittal will enhance surveillances in an effort to reduce the probability of an asymmetric rod withdrawal event during routine operation. In addition, they will increase the awareness of the technical staff of the potential for an asymmetric rod withdrawal event, as well as provide appropriate guidance to the Operations staff. The commitments being made by this submittal will further reduce the probability of an asymmetric rod withdrawal event during routine operations. As such, the Authority is addressing this issue by taking appropriate measures to enhance both personnel and equipment performance.

ATTACHMENT III TO IPN-93-109

COMMITMENTS MADE IN THE AUTHORITY'S SEPTEMBER 21, 1993 SUBMITTAL

RELATED TO

SINGLE FAILURE VULNERABILITY OF WESTINGHOUSE ROD CONTROL SYSTEM

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Commitment Number	Commitment	Due Date
93-109-01	For both insertion and withdrawal, the lift, movable and stationary currents for each rod will be monitored in the power cabinets (one time surveillance).	Prior to restart
93-109-02	The surveillance specified in commitment number 93-109-01 will be repeated following the performance of extensive maintenance and/or modification of the Control Rod Drive Control System, or repairs/replacement of the power supply, for all rods affected.	To be implemented after restart as applicable
93-109-03	Evaluate the appropriateness of modifying the Rod Control System current order timing to prevent an uncontrolled asymmetric rod withdrawal.	120 days after Westinghouse provides details of proposed mod.
93-109-04	Based on evaluation (see commitment 93-109-03) advise NRC of long term corrective actions.	180 days after Westinghouse provides details of proposed mod.

ATTACHMENT IV TO IPN-93-109

COMMITMENTS MADE IN THE AUTHORITY'S AUGUST 5, 1993 SUBMITTAL

RELATED TO

SINGLE FAILURE VULNERABILITY OF WESTINGHOUSE ROD CONTROL SYSTEM

NEW YORK POWER AUTHORITY
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Commitment Number	Commitment	Due Date
93-094-01[1]	Submit 90 day response addressing GDC 25	September 20, 1993
93-094-02	Review applicability of any additional information and re-evaluate appropriateness of commitments made in IPN-93-094.	To be determined if additional information is received.
93-094-03	Revise procedures pertaining to the rod control system to include additional guidance to operations, maintenance and I&C personnel.	Prior to restart
93-094-04	Revise procedures pertaining to the rod control system to include precautionary measures to verify the correct operation of the system.	Prior to restart
93-094-05	Revise procedures to require the exercising of each bank of RCCAs to ensure the correct motion of the rods following the performance of maintenance on the control rod drive system.	Prior to restart
93-094-06	Verify functionality of rod deviation alarm with simulated signals.	Prior to restart
93-094-07	Verify functionality of the rod deviation alarms with simulated signals monthly.	This is a current procedural requirement
93-094-08	Perform RCCA operability test every 31 days.	This is a current procedural requirement
93-094-09	Exercise each RCCA bank individually prior to withdrawing the shutdown bank and bringing the reactor critical.	Prior to restart
93-094-10	Train licensed reactor operators on rod control system, including a discussion of the malfunctions at Salem and appropriate operator response in the event of a rod control system malfunction.	Prior to restart
93-094-11	Inform I&C staff and associated systems engineers of Salem event.	Prior to restart

[1]Complete with this letter.

93-094-12	Review and revise, as necessary, abnormal event response procedures to assure adequate guidance to operators in the event of a rod control system malfunction.	Prior to restart
93-094-13[2]	Review WCAP-13803 and applicability of generic analysis to Indian Point 3.	Within two weeks of obtaining WCAP-13803

[2]Completed