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NewYorkPower Authority

May 17, 1985
IPN-85-26

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactor Branch No. 1
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Additional Information Regarding Generic Letter 83-28
"Required Actions Based on Generic Implications of the
Salem ATWS Events"

Dear Sir:

Attachment A to this letter provides the Authority's response to your letter dated March 7, 1985. That letter requested additional information relating to Indian Point 3's safety-related equipment classification, vendor interface program and post-maintenance testing.

If you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons
John C. Brons
Senior Vice President
Nuclear Generation

cc: Resident Inspector's Office
Indian Point Unit 3
U. S. Nuclear Regulatory Commission
P. O. Box 66
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ATTACHMENT A

GENERIC LETTER 83-28
Response to 3/7/85 Request for Information
Items 2.1, 2.2.1, 2.2.2, 3.1.3,
3.2.3 and 4.5.3

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

ATTACHMENT A

Response to 3/7/85 Request for Information
Concerning Generic Letter 83-28

Item 2.1 (part 2)

Licensee needs to submit detailed information describing his vendor interface program for reactor trip system components. Information supplied should state how the program assures that vendor technical information is kept complete, current and controlled throughout the life of the plant and should also indicate how the program will be implemented at INDIAN POINT 3.

Response

The Authority's response to Item 2.1 (part 2) was contained in the Authority's response to Item 2.2.2 which was submitted to the NRC by letter dated July 3, 1984 (IPN-84-22).

As stated in that letter, Westinghouse, the NSSS vendor for Indian Point 3, has instituted a return receipt system to ensure that Westinghouse Technical Bulletins pertaining to the Reactor Trip System in addition to other safety-related equipment are received by the utilities. The Authority's Administrative Procedure AP-37, Rev. 6 "Feedback of Operating Experience to Plant Staff" describes the review process of the technical bulletins.

With respect to non-NSSS vendors, the Authority supports enhancing industry communications through the Vendor Equipment Technical Information Program (VETIP) by using the INPO NPRDS and SEE-IN programs. Nuclear Generation Procedure 5, Rev. 2 "Operating Experience Review Program" describes the Authority's interface with the SEE-IN program. Nuclear Generation Procedure 15, Rev. 2 "Conduct of Operational Analysis and Training" and Plant Procedure PFM-6, Rev. 0, discuss the applicability and control of NPRDS.

Item 2.2.1

Licensee needs to submit information on how equipment will be classified as safety-related and will be designated as such on plant documentation as requested in sub-item 2.2.1.1

Response

The Authority has, historically, classified its safety-related equipment on a system level. The Authority continues to maintain that this approach is conservative. However, the Authority has undertaken a major effort to develop a safety-related component list. A bid specification describing the scope of work has recently been issued. The completion of this item is expected by mid-1986.

In addition, it should be noted that some drawings and equipment specifications are indicated as safety-related (i.e., seismic category or Class A as originally defined during the licensing of IP-3). Appropriate drawings and specifications will be reviewed, subsequent to the completion of the classification effort, to verify whether or not they are indicated as safety-related. Also, following completion of this effort, the Authority will evaluate the need and practicality of physically designating additional drawings, specifications and other pertinent plant documentation as safety-related.

The criteria for classifying safety-related equipment as stated in the aforementioned bid specification are as follows:

Safety-related structures, systems and components are those relied upon to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. In addition, safety-related components, structures and systems are required to assure 1) integrity of reactor coolant pressure boundary, 2) capability to achieve and maintain safe shutdown, and 3) capability to prevent or mitigate the consequences of postulated accidents which could result in potential off site exposures comparable to the guidelines of 10 CFR part 100.

Item 2.2.2

Licensee needs to present his evaluation of the NUTAC program and describe how it will be implemented at INDIAN POINT 3. The staff found the NUTAC program fails to address the concern about establishing and maintaining an interface between all vendors of safety-related equipment and the utility. Accordingly, the licensee will need to supplement his response to address this concern. This additional information should describe how current procedures will be modified and new ones initiated to meet each element of item 2.2.2 concern.

Response:

The Authority still considers that the Vendor Equipment Technical Information Program (VETIP) as defined in the March 1984 NUTAC document is a valid response to Item 2.2.2 of the NRC Generic Letter 83-28. The Authority is implementing the program as described therein. Accordingly, it is requested that NRC reanalyze and reconsider the request for additional information.

In addition, following the issuance of the generic letter, the Authority has revised, developed and implemented procedures to address adequately the concerns of Item 2.2.2 related to controls of vendor technical information.

Item 3.1.3

Results of review of test and maintenance programs shall identify any post-maintanenence testing that may degrade rather than enhance safety and shall describe actions to be taken including submitting needed Technical Specification changes.

Response

Review of incoming vendor technical information and engineering recommendations will continue to be performed. In the event that the potential for degradation of RTS safety due to post-maintenance test requirements is identified in the future, appropriate changes and associated justifications will be submitted to the NRC for approval. To date no post-maintenance testing which would degrade safety has been identified.

Currently, the Authority is reviewing, for plant specific applicability, the NRC SER on WCAP-10271, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation Systems."

Item 3.2.3

Same as item 3.1.3.

Response:

By letter dated September 7, 1984, the Authority has submitted proposed changes to the Technical Specifications relating to "Station Batteries." One of the proposed changes (i.e., equalizing charge frequency period) pertained to a recommendation made by the manufacturer and also described in a related industry standard (IEEE-450-1980). This proposed change is an example of one made to decrease the possibility of degrading the margin of safety.

The Authority will continue to review and propose changes related to post-maintenance testing requirements when and if identified.

Item 4.5.3

Licensee needs to submit a description of the specific implementation plan for INDIAN POINT 3 after NRC reviews the WCAP-10271 and supplement 1.

Response

As stated in response to Item 3.1.3, the Authority is currently reviewing the NRC SER pertaining to WCAP-10271, Supplement 1. The Authority has, also, recently received guidelines for requesting revisions to reactor protection system technical specifications from the Westinghouse Owners Group (WOG). The Authority will review the NRC SER and the WOG guidelines and take appropriate actions, including proposing technical specification changes, if necessary.