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NewYorkPower Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

October 26, 1983
IPN-83-89

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

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
NUREG-0737; Item II.B.3
Post-Accident Sampling System

Dear Sir:

This letter and its Attachment serve to respond to the request for information contained in Mr. S. A. Varga's letter dated August 15, 1983 regarding the Indian Point 3 post-accident sampling (PAS) system. Mr. Varga's letter transmitted to the Authority the NRC Staff's Safety Evaluation Report (SER) for the PAS system and stated that four of the eleven criteria had not been fully resolved. The Attachment to this letter contains additional information concerning two of the four 'open' criteria and a schedule for responding to the remaining requests.

The Authority notes that the Confirmatory Order dated March 18, 1983 indicates that the subject NUREG-0737 item has been completed for Indian Point 3. Such a status was based on physical installation and testing of the PAS system and could not have reflected the NRC staff's considerations addressed in the SER attached to Mr. Varga's letter. The Authority considers NUREG-0737, Item II.B.3 to be complete to the extent that NRC concerns identified in the aforementioned SER have been resolved.

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

Very truly yours,

J. P. Bayne
Executive Vice President
Nuclear Generation

cc: Resident Inspector's Office
Indan Point Unit 3
U. S. Nuclear Regulatory Commission
P. O. Box 66
Buchanan, New York 10511

State of New York
County of Westchester

Subscribed and sworn to before me this 26 day of October 1983

Jeanne La Luna
Notary Public

JEANNE LA LUNA
NOTARY PUBLIC, STATE OF NEW YORK
NO. 60-4614305
QUALIFIED IN WESTCHESTER COUNTY
TERM EXPIRES MARCH 30th 19 85

Attachment to IPN-83-89
Supplemental Information Regarding
Item II.B.2 of NUREG-0737

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three-hour frame established above, quantification of the following:

(a) Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and non-volatile isotopes):

Clarification: 2(a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non-volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Ce, ^{85}Kr, ^{140}Ba, and ^{88}Kr, (see Vol. II, Part 2, pp. 524-527 of Rogivin Report for further information).

2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature date and sample location.

Response to Criterion/Clarification 2(a).2

An interim procedure required for relating radionuclide concentrations and physical parameters to core damage is presently being developed and will be in place by December 1, 1983. A final procedure will be implemented subsequent to Authority review of the Westinghouse Owner's Group generic procedure scheduled for issuance in the first quarter of 1984.

Criterion: (3) Reactor coolant and containment atmosphere sampling during the post accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

Clarification: (3) System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

<u>Response to Criterion /Clarification 3</u>

The Authority's letter IPN-83-38, dated May 10, 1983 provided a response to the above cited criterion and clarification and stated that the environmental qualification of PAS System valves which would not be accessible after an accident was being evaluated as part of the Environmental Qualification Program. It is the Authority's intention to replace or qualify the PAS system valves inside containment during the mid-cycle steam generator inspection outage, expected to begin in November, 1984.

Criterion: (10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Response to Criterion 10

The post-accident sampling (PAS) system at Indian Point 3 was designed in accordance with the guidance contained in NUREG-0578 and NUREG-0737. The system was installed and operational well in advance of the NRC's issuance of this "clarification" which, in effect, constitutes a significant change in the system design requirements. By letter dated May 10, 1983, IPN-83-38, the Authority committed to review the PAS system versus the requirements of Regulatory Guide 1.97, Revision 2. The schedule for implemementation of Revision 2 to the Guide is given in the Authority's response to Supplement 1 to NUREG-0737, letter IPN-83-26 dated April 18, 1983. The first phase of this effort will establish the degree of compliance with the Guide and is expected to be completed during 1983. The second phase of the program, if necesary, will involve the determination of specific justification of exceptions to the Guide and the development of specific design modifications, if required. It is expected that any required justifications, assessments and (or) corrective actions will be defined by the second quarter of 1984. The implementation schedule for any modifications will also be established at that time.

The Authority will compile accuracy, range, and sensitivity information for the existing PAS system, not including the "matrix" identified in the clarification to criterion 10, and submit such information to the NRC by November 30, 1983.

Chemistry personnel have been trained in the use of PAS system procedures. Refresher training will be held on an annual basis, consistent with the guidelines of NUREG-0660 for operator training. PAS system analytical equipment is used for routine chemical analyses and hence are calibrated or checked for accuracy more frequently than once every six months.

Criterion: (11) In the design of post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification II.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close oncontainment isolation or safety injection signals.

Clarification: (11)(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

Response to Criterion 11

The Authority's letter IPN-83-38 dated May 10, 1983 provided a response to the above cited clarification and criteria. Additional information is provided below in response to the SER attached to Mr. S. A. Varga's letter dated August 15, 1983.

The PAS system has provisions for purging of the sample lines to assure that a representative sample is obtained for analysis. Heat tracings will be added to the containment atmosphere sample lines during the next (cycle 4/5) refueling outage, Issues related to plate-out in the sample lines will be addressed in the November 30, 1983 submittal. The PAS system sample lines employ two in-series containment isolation valves to limit potential reactor coolant loss from a rupture of the sample line.