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New York Power Authority

March 4, 1997
JPN-97-008
IPN-97-029

Ms. Mary Drouin
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
MS T-10 E50
Washington, DC 20555-0001

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Comments on Draft NUREG-1560

Reference: Draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance."

Dear Ms. Drouin:

Enclosed for your consideration are the Authority's comments on the referenced document pursuant to the November 14, 1996 Federal Register notice. The comments are divided into three groups: editorial comments; technical comments of a general nature; and comments specific to the Indian Point 3 and James A. FitzPatrick Nuclear Power Plants.

The Authority appreciates this extensive undertaking on the part of the NRC staff to develop a comparison of utility's Individual Plant Examination (IPE) programs, and characterization of the attributes of a quality Probabilistic Risk Assessment. We expect to make good use of NUREG-1560 in updating the IPEs of NYPA's nuclear plants, and in demonstrating their viability for use as a basis for safety-related decision making.

If you have any questions, please contact Ms. C. D. Faison.

Very truly yours,

James Knubel
Chief Nuclear Officer

Attachment: as stated
cc: next page

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Attachment to JPN-97-008 / IPN-97-029

New York Power Authority Comments

Individual Plant Examination Program:
Perspectives on Reactor Safety and Plant Performance
Draft NUREG-1560

Editorial Comments

1. Suggest that:

    - The summary (Volume 1) be abbreviated to omit much of the detail presented in Volume 2

    - Within Parts 2 and 3 of the report, conclusions that apply to all BWRs or all PWRs not be repeated for specific types of BWRs or PWRs.

2. The page numbers in the index listed for Section 9 are incorrect.

3. The New York Power Authority's James A. FitzPatrick Nuclear Power Plant is usually referred to as JAF. Though FitzPatrick is a satisfactory alternative, Fitzpatrick (small "p") is incorrect.

4. Page 9-28. The statement, 4 lines from the bottom, that "The Indian Point 2 and 3 submittal ....." implies that a joint submittal was made. It should read "The Indian Point 2 and 3 submittals ...."

5. All sources of information in the document should be referenced; e.g., the deliberations of the 1995 Steam Explosion Review Group described on page 14-45.

General Technical Comments

1. On page 14-19, it is asserted that a quality PRA justification will be provided for excluding failure modes. Elaborate consideration of all failure modes can be an unwieldy task. Perhaps this could be redrafted to require justification for excluding any failure or failure mode with a probability of greater than 0.01 of the probability of most probable failure mode or failure of that equipment item.

Comments Specific to the Indian Point Unit 3 Nuclear Power Plant

1. Pages 9-26, 9-28 and 9-29. Subsequent to submittal of the IPE, the following plant improvements have been made at IP3 in response to issues raised in the IPE and elsewhere:

    - The revision of emergency operating procedure FR-H.1 to provide explicit guidance to the operators for restoring the flow of main feedwater should a loss of secondary heat sink be indicated.

    - The provision of adequate seismic support to the portion of the fire protection piping deluge valve station located in the control room at the 15-ft elevation and the installation of a waterproof door to the deluge valve station room.

    - The revision of I & C periodic calibration procedures to address repeated maintenance-induced failures of the emergency diesel generator exhaust fan temperature controllers.

    - The installation of RTDs to provide a control room alarm upon detection of nigh ambient temperatures on the 15 ft. and 33 ft. elevations of the control building.

    - The revision of maintenance procedures for the auxiliary boiler feedpump building exhaust fans, louvers and dampers. This revision will improve the availability of the AFW pump room ventilation system.

    - The expansion in scope of the diesel generator functional tests to verify that the exhaust fan auto-start controls operate.

2. Pages 3-75 and 11-132. At the time the IP3 IPE was submitted, IP3 operated with PORV block valves closed. However, it should be noted that the PORVs were replaced during a recent outage, are not experiencing leakage, and are in service with the block valves open.

Comments Specific to the James A. FitzPatrick Nuclear Power Plant

1. Pages 9-9, 9-10 and 9-23. The procedure modifications and training for the use of firewater as a backup to the RHRSW have been implemented.

2. Page 9-23. The RCIC room exhaust fans are now powered from dc power sources. Other plant improvements made that will further reduce risk are:

   - The installation of bonnet vents on the LPCI and core spray injection valves to preclude the potential common-cause pressure locking of the valves.

   - The modification of the fire protection system to provide EDG jacket cooling water directly through the cross tie to the ESW system. The use of this cross tie is addressed in an operating procedure.

   - The installation of a keylock bypass switch that allows LPCI and core spray injection valves to be manually opened from the control room.

   - The installation of a keylock bypass switch to allow HPCI auto-transfer on high suppression pool level to be bypassed from the control room rather than by lifting leads in a relay room panel.

   - The change of RHR minimum flow bypass valves from normally closed to normally open.

   - The installation of switches to permit transfer to the alternate power supply for LPCI injection valves to be made from the control room.

3. Pages 8-17, 9-33 and 9-38. A hardened vent was not installed at JAF. Rather, the New York Power Authority determined that the existing hardened wetwell vent path meets the hardened vent design criteria or their intent. The NRC staff concurred with this determination and noted that plant procedures and training are adequate to provide the information and guidance necessary to make effective use of the hardened wetwell vent capability. The NRC staff concluded that the existing wetwell vent was adequate (NRC Docket No. 50-333, SER dated September 28, 1992).

4. Page 11-26. In Table 11.9 it is stated that the characterization of SBO sequences as causing core damage in the short or long term was based on sequences comprising only 90 % of CDF. It would be more correct to say the characterization was based on eight sequences comprising at least 90 % of CDF.

5. Page 12-13. The conditional probabilities of early containment failure at FitzPatrick are high because a station blackout is the dominant cause of core damage. A station blackout precludes flooding of the drywell floor.

6. Pages 13-16 and 13-17. Figure 13.1 and the accompanying text show and comment on a low HEP assigned for a failure to depressurize during transients. This value (1.4×10^{-4}) was justified because 50 minutes are available in which to depressurize the reactor and

because of the intensive training given for this action. Under normal conditions, the EOPs would instruct the operator to override ADS. If the operator fails to override ADS, ADS will automatically depressurize the RPV. If the operator overrides ADS, then there is an increased probability that the operator has successfully diagnosed the event.

7. Pages 13-19 and 13-20. Figure 13.3 and the accompanying text show and comment on a 10^{-5} HEP assigned to a failure to inhibit ADS in ATWS scenarios. As noted in Sections E2.1.2 and E2.2.2 of the IPE, this probability was used because the action was believed to be an immediate action performed automatically by the operators when an ATWS occurs. The EOP for ATWS event (EOP-3) has an instruction to override ADS in both reactor power and RPV water level branches in the procedure. In addition, on an automatic ADS actuation signal, an alarm sounds in the control room which gives the operator two minutes to override ADS before automatic depressurization. In view of the training and simulator exercises that the operator has on this event and the multiple operator actions that are required to fail, a 10^{-5} probability was judged to be valid. This probability value had also been independently reviewed and concurred by a renowned human reliability analysis consultant before the IPE was submitted.