123 Main Street
White Plains, New York 10601
914 681.6240

New York Power Authority

John C. Brons
Executive Vice President
Nuclear Generation

January 12, 1988
JPN-88-001
IPN-88-001

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

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Revisions to Comments on Draft NUREG-1150
Reactor Risk Reference Document

Reference: 1. NYPA letter, J. C. Brons to NRC dated September 28, 1987 (JPN-87-051, IPN-87-045), regarding NYPA comments on Draft NUREG-1150.

Dear Sir:

In Reference 1, the Authority submitted comments on Draft NUREG-1150, "Reactor Risk Reference Document". The purpose of this letter is to correct editorial errors in the Authority's comments. The enclosed revised pages supersede and replace in their entirety pages 8, 14, 31 and 48, in Attachment 1 of Reference 1. A bold vertical line has been drawn in the margin adjacent to the portion of the text actually changed.

If you have any questions regarding this matter, please contact Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

John C. Brons

John C. Brons
Executive Vice President
Nuclear Generation

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Enclosures

cc: U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Office of the Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 136
Lycoming, New York 13093

Mr. H. Abelson, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
U. S. Nuclear Regulatory Commission
7920 Norfalk Avenue
Bethesda, Maryland 20014

Resident Inspector's Office
Indian Point 3
U. S. Nuclear Regulatory Commission
P. O. Box 377
Buchanan, New York 10511

Joseph D. Neighbors, Sr. Project Manager
Project Direcorate I-1
Division of Reactor Projects - I/II
U. S. Nuclear Regulatory Commission
7920 Norfalk Avenue
Bethesda, Maryland 20014

ATTACHMENT I
TO
JPN-88-001
IPN-88-001

REVISION TO COMMENTS ON DRAFT NUREG-1150 REACTOR RISK REFERENCE DOCUMENT

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333 DPR-59
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286 DPR-64

pool heat removal capacity and operator actions to control water level during an ATWS event. STCP improvements permit plant-specific design features and operator actions to be accurately modeled.

1.3.7. Pressurized Water Reactors

In NUREG-1150, direct containment heating is a major contributor to uncertainty for plants with large, dry containments. Early containment failure as a result of direct heating requires that two conditions coexist: high reactor pressure coincident with reactor vessel bottom failure due to melting core. Authority analyses show that these two conditions are highly unlikely to occur together. Rather, other primary system locations (like the hot leg) are likely to fail before the vessel. This rapidly depressurizes the reactor removing one of the two early containment failure prerequisites. Uncertainty associated with large dry containments should be reduced to reflect this.

1.3.8. Emergency Preparedness Models

Early fatality calculations are very sensitive to the models used for emergency response. NUREG-1150's assumption that five percent of the population does not respond to emergency notification is the determining factor for early fatalities. This technical basis for this assumption should be re-examined and the value re-established.

the boiling water reactor. Nevertheless, the MARCH3 code does not realistically analyze boiling water reactors. Specifically, the following problem areas have been identified.

2.2.1.2.1. Treatment of the Reactor System as One Volume

As a carryover from the PWR calculation methodology, the BWR reactor system is modeled as a single node saturated system. The net effect is an overprediction of the steaming rate and, therefore, the increased transport of fission products away from the reactor core. In reality, there can be considerable subcooling in the downcomer and lower plenum regions of the reactor vessel. The STCP should be modified to model BWRs as multi-volume systems.

2.2.1.2.2. Problems Associated with the Treatment of Emergency Core Coding (ECC) Systems

Two problems have been identified. The first is the manner in which ECCS is assumed to be initiated. In a BWR, initiation of the ECCS pumps as well as the ADS system is determined by the reactor vessel downcomer water level. The Source Term Code Package does not treat the water level in the reactor system in a realistic manner and, therefore, is unable to relate the initiation of the ECCS and ADS systems to downcomer water level. The STCP finds an "equivalent" water level and does not account for different water temperatures in the reactor vessel. Therefore, the analyst must input the time at which the ECCS is initiated and is turned off. Similarly, the analyst must input

Figures TC1.3 and TC1.4. Steaming pressurizes the drywell again due to suppression pool heatup and the passage of non-condensibles from the core to the suppression pool and finally to the drywell. This recloses the ADS valves and the reactor system begins to repressurize. The drywell pressure again begins to decrease, but not fast enough to reopen the ADS valves before the reactor vessel fails. At 225 minutes the core slumps into the lower head, followed almost immediately by lower head failure. Because the reactor system is at high pressure where the reactor vessel fails, the reactor system depressurization causes the drywell pressure to rapidly exceed the containment failure pressure, and the containment fails.

Although the reactor vessel has failed in the RMA analysis, the CRD pumps would continue to deliver water, which ends up in the reactor cavity. The flow rate of water is sufficient to partially quench the core debris. This delays significant core/concrete interaction until the inventory of the condensate storage tank is depleted and the CRD pumps are assumed to trip. Both the RMA and STCP analyses do not assume that operator action to replenish the condensate storage tank is taken. The probability of such an action being taken is actually very high.

The RMA results differ significantly from those produced by the STCP. The largest contributors to these differences are the assumptions dealing with uninterrupted delivery of water by the

In the RMA analysis, the cesium hydroxide is readily removed by chemisorption processes onto stainless steel surfaces because of the relatively long residence time in the reactor system during the period of time between 200 and 250 minutes. A similar effect is observed in the behavior of tellurium. The results presented in Table 3.11 show that a moderate portion of cesium iodide is retained in the reactor system, whereas cesium hydroxide is removed in the reactor coolant system to a much large extent. Consequently, the majority of cesium iodide ends up in the suppression pool, whereas only a small portion of cesium hydroxide and a very small portion of tellurium ends up in the suppression pool water.

The RMA analysis of the TC3 sequence shows that the thermal-hydraulic behavior of the reactor system is not dominated by artificially induced flow rates. Therefore, adequate residence times exist for the chemisorption processes to remove cesium hydroxide and tellurium. The behavior of cesium iodide is affected by the thermal behavior of the reactor system and highly influenced by the flow behavior following core slump and collapse.

In view of the fact that the STCP calculation for the TC3 sequence predicts containment venting prior to vessel head failure, it is not surprising that Table 3.9 shows that the suppression pool plays a major role in reducing the source term to the small quantities shown. As discussed earlier, early containment venting is an error and, therefore, the results in Table 3.9 are also in error. These STCP errors overpredict