Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511
914-736-8000

New York Power Authority

March 20, 1996
IPN-96-032

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

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Additional Commitment Implementation Assessment Results

Reference: 1. NYPA letter (IPN-94-054), W. A. Josiger to the NRC, dated May 2, 1994, "Commitment Implementation Assessment."

Dear Sir:

The New York Power Authority (NYPA) reviewed the implementation of commitments that NYPA made to the Nuclear Regulatory Commission as part of the Restart and Continuous Improvement Plan (RCIP), and identified certain items that required clarification or revision. Also, in the course of conducting normal business, additional commitments were identified as requiring clarification or changes from previous submittals. Reference 1 provided information regarding some of the items we identified as needing revision. This letter notifies you of other commitments that were identified, the changes or clarifications to those commitments, and their current status.

Attachment I provides a discussion of the commitments that require clarification or revision, the changes or clarifications, and status, as well as the source of the commitment. Attachment II lists NYPA's commitments being made by this submittal. If you have any questions, please contact Mr. K. Peters.

Very truly yours,

Robert J. Barrett

Robert J. Barrett
Plant Manager
Indian Point 3 Nuclear Power Plant

Attachments
cc: See next page

A001

cc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. George F. Wunder,
Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14 B2
Washington, DC 20555

EXECUTIVE SUMMARY

Reviews were performed of the implementation of commitments that NYPA made to the Nuclear Regulatory Commission as part of the Restart and Continuous Improvement Plan (RCIP). Also, in the course of performing other work, additional discrepancies were identified. The following are discrepancies identified that are described in further detail in the attachment:

A. The program to meet the requirements and previous commitments associated with NUREG-0578 Item 2.1.6.a/NUREG-0737 Action Item III.D.1.1, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs," did not include an applicable portion of a system and provide a test of a previously identified portion of a system.

B. NYPA revised a commitment previously provided in a response to Generic Letter 91-06 dated April 29, 1991, entitled "Resolution of Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies, Pursuant to 10 CFR 50.54(f)." The revised action is a result of a change in the frequency of Nuclear Plant Operators' (NPO) tours conducted for monitoring the status of safety related DC power supplies from the frequency provided in response to Generic Letter 91-06.

C. The actual range of the Main Steam Line radiation monitors is in accordance with requirements of NUREG-0737 and not Regulatory Guide 1.97 as stated in the FSAR. NYPA also discovered the range listed in the FSAR Table 11.2-7 does not reflect the value provided in responses to NUREG-0737 Item II.F.1 and Regulatory Guide 1.97. The range value listed in the FSAR only reflects a local analog display and not the digital display capability provided locally and in the control room to meet NUREG-0737 and Regulatory Guide 1.97.

D. Additional procedural changes were not implemented for NUREG-0737 Action Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used In a Postulated Accident." Procedures were to be revised to ensure that two valves in the containment spray system would be operated prior to high head recirculation to meet dose limits and that the NRC was informed if the final dose analysis altered the modifications stated in the response.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

A. PROGRAM TO REDUCE LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

Recommendation 2.1.6.a of NUREG-0578 required establishing a program to identify and reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious accident. Clarifications to the NUREG-0578 recommendations were provided by NUREG-0737 Action Item III.D.1.1. The Indian Point 3 license was revised by Amendment 38 dated October 7, 1981, to contain License Condition 2.L for a leakage reduction program reflecting NYPA's submittal. The program and tests were implemented, and found acceptable by the NRC. NYPA Audit 93-20 verified program compliance to commitments and identified the following weaknesses and nonconformances.

Finding Audit 93-20

1. **Commitment**

NYPA provided a revised commitment for NUREG-0578, Item 2.1.6.a, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs." NYPA's response stated that it "has established a program to identify and reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident." The systems that were identified as part of the program for leak identification and reduction included, "the CVCS Volume Control Tank (VCT) up to the outlet isolation valve including gas space."

2. **Source**

NYPA's supplemental response to NUREG-0578 Item 2.1.6.a, dated February 3, 1980 (IPN-80-15), revised the response provided by letter dated January 8, 1980. NYPA's response to NUREG-0737 Item III.D.1.1, was stated to be completed in letter dated December 30, 1980.

3. **Finding (Discrepancy)**

As a result of QA Audit 93-20, dated July 30, 1993, Corrective Action Request (CAR) No. 864 was issued identifying the following nonconformances:

1) The gaseous space of the VCT up to the outlet isolation valve was not included in the program and there was no test procedure for leak testing (gases).

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

A. PROGRAM TO REDUCE LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

3. **Finding (Discrepancy) cont'd**

2) The Reactor Coolant Pump seal return line to the VCT was not included in the program but is applicable to the program since it may not be isolated for a SBLOCA although isolated on a Phase B containment isolation.

3) Program procedure AP-35 Attachment 1 does not include test procedure 3PT-R10A, RHR System Integrity Test.

4) An Operating Experience Review Group (OERG) recommendation (RIND No. 92273) regarding revised lesson plans for training, remained open after issuance of LER 92-005 dated May 20, 1992.

4. **Status**

The following corrective actions for the findings were implemented:

1) Program procedure AP-35, "Integrity of Systems Outside Containment," was changed by Revision 6 dated August 13, 1993, to include the RCP seal return line. Procedure 3PT-R72, "Volume Control Tank Integrity Test," was changed by Revision 5 dated September 14, 1993, to require gaseous leak testing for the VCT up to the outlet isolation valve.

2) Program procedure AP-35, "Integrity of Systems Outside Containment," was changed by Revision 6 dated August 13, 1993, to include the RCP seal return line to the VCT.

3) Procedure AP-35 Attachment 1 and Procedure 3PT-C01, "Total Leakage Rate Monitoring Tabulation," were changed to include 3PT-R10A, "RHR System Integrity Test."

4) The Training Department evaluated the OERG Department recommendation, tracked in the corrective action system at that time as RIND No. 92273, and determined that revised lesson plans for training were not required and the recommendation was closed.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

A. PROGRAM TO REDUCE LEAKAGE FROM SYSTEMS OUTSIDE CONTAINMENT

4. **Status (cont'd)**

The original verification of NYPA's implementation of TMI Action Item III.D.1.1 was documented by NRC Inspection Report 50-286/81-02 dated August 13, 1981. The inspector reviewed program procedure AP-35 and concluded that tests 3PT-M16, "Safety Injection System," and 3PT-M18, "Residual Heat Removal System" were acceptable. The inspector noted four additional refueling tests not available at the time and identified open item 81-02-04 on the issue. A followup inspection (Inspection Report 50-286/82-10 dated July 30, 1982), closed unresolved item 286/81-02-04 based on a detailed review of the tests used to meet TMI Action Item II.D.1.1. However, changes have occurred since the item closeout. Research identified additional discrepancies between the commitments provided to the NRC in response to NUREG-0578 Item 2.1.6.a , NUREG-0737 Action Item III.D.1.1 and current documentation. FSAR Section 6.2.3 and the program (AP-35) identify two additional systems; the BIT and the Containment Hydrogen Monitoring System. Although the BIT is part of the Safety Injection System which was originally identified as part of the program, the BIT was not specifically identified as part of the program nor was the Containment Hydrogen Monitoring System. Also, the tests listed for each system originally identified as part of the program did not include 3PT-R127 for the BIT nor 3PT-R69A&B for the Containment Hydrogen Monitoring System. This issue is considered closed.

B. ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES

In Generic Letter 91-06, "Adequacy of Safety-Related DC Power Supplies," the NRC requested that licensees provide information about their DC power supplies. NYPA provided its response and noted that existing design features, administrative controls, and surveillance testing provide reasonable alternatives to NRC recommendations. However, subsequent to the Generic Letter submittal, it was discovered that the stated inspection interval for equipment status was incorrect. The following description clarifies the differences:

1. **Commitment**

NYPA's response to Generic Letter 91-06 stated that Nuclear Plant Operators (NPO) perform tours at a frequency of four hours to verify proper battery charger output levels (current and voltage) for Generic Letter questions 5 and 9. The response also stated that the tours provided reliable monitoring of battery charger status independent of the "Battery Charger Trouble Alarm" circuit.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

B. ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES (cont'd)

2. **Source of Commitment**

NYPA letter, R. E. Beedle to the NRC, dated October 28, 1991 (IPN-91-039), Response to Generic Letter 91-06.

3. **Finding (Discrepancy)**

The procedure that controlled the monitoring and recording of plant operating parameters and conditions at the time, Operating Directive (OD)-5, "Log Keeping," contained conventional log sheets which were not in accordance with commitments contained in the response to Generic Letter 91-06 since the log required the NPO to perform a tour frequency for monitoring battery charger output twice per shift instead of at a frequency of every four hours. The current conventional log sheets are reformatted and referenced in procedure OD-3, "Operator Rounds and Log Sheets," as Operations Periodic Task (OPT) sheets. OPT-16, "Conventional Hot Log Sheet," and OPT-17, "Conventional Cold Log Sheet," contain the requirement for monitoring and recording of battery charger output once a shift.

4. **Status**

At the time the response was developed, NPO tours were typically conducted twice per shift, not specifically every four hours as stated in the generic letter response. The procedure that provided instructions for the monitoring and recording of plant operating parameters and conditions was Operating Directive (OD)-5, "Log Keeping." OD-5 at the time included Conventional Log Sheets which contained the requirement to verify proper battery charger output levels (voltage/current). OD-5 was revised in February 1993 to change the NPO tour frequency for monitoring battery charger output to once per shift. Subsequently, the log sheets from procedure OD-5 were reformatted and relocated to procedure OD-3, "Operator Rounds and Log Sheets," as a reference (OPT-16 and OPT-17). The current Conventional Log Sheets (OPT-16 (Hot) and OPT-17 (Cold)) requires monitoring and recording of battery charger current and voltage once per shift.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

B. ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES (cont'd)

4. **Status (cont'd)**

The change in the NPO tour frequency to once per shift resulted from an assessment of operator activities that concluded that NPO tours at a frequency of twice a shift consumed so much time that not enough time was devoted to higher priority duties. NPO verification of the local alarm status of the battery chargers serves as a backup indication of the operation of the chargers. Problems with any one of the battery chargers will be indicated in the control room by the "Battery Charger Trouble Alarm," and/or existence of abnormal DC Bus voltage levels on the selectable voltmeter in the control room. The DC Bus voltage is monitored twice a shift in accordance with Control Room Log sheet OPT-11 (Hot) and OPT-12 (Cold). Existing ARP's and ONOP's provide the operators with guidelines for responding to alarms or abnormal conditions. The frequency for conducting NPO tours for the battery chargers is considered acceptable based on industry operating experience related to equipment unavailability and detecting declining performance trends.

C. RANGE OF THE MAIN STEAM LINE RADIATION MONITOR

Post-accident monitoring and range requirements for the Main Steam Line (MSL) radiation monitors (R-62A-D) were contained in NUREG-0578 Recommendation 2.1.8.a, NUREG-0737 Item II.F.1.1, and Regulatory Guide 1.97 Table 3. The *numerical* range requirements for NUREG-0737 and Regulatory Guide 1.97 were equivalent (i.e., 1E-1 to 1E3). However, the maximum range *basis* values for NUREG-0578 and NUREG-0737 were **Xe-133 equivalent**, whereas Regulatory Guide 1.97 required the range value to be based on a **radionuclide mix**. The Authority's response to NUREG-0737 Item II.F.1 committed to provide monitoring that met the NUREG-0737 range requirements (i.e., 1E-1 to 1E3 microcuries per cubic centimeters (uCi/cc) based on Xe-133 equivalent). NYPA's subsequent commitment to Regulatory Guide 1.97 range requirements for this potential release path (Main Steam Line) was to meet the Regulatory Guide 1.97 range requirement. FSAR Section 11.2 states the monitors meet the requirements of Regulatory Guide 1.97 and Table 11.2-7 lists a range for channel R-62 of 1E-4 to 1E+1 uCi/cc. However, although the numerical values in NUREG-0737 and Regulatory Guide 1.97 were the same (i.e., 1E-1 to 1E3 uCi/cc), their basis was different and therefore the capabilities to meet them are different. Also, the range listed in the FSAR does not correlate with range commitments. The following is a clarification of the commitment for the range of the Main Steam Line monitors (Channel R-62).

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

C. RANGE OF THE MAIN STEAM LINE RADIATION MONITOR (cont'd)

1. **Commitment**

NYPA would provide radiation monitors for the main steam lines that met the requirements of NUREG-0737 Item II.F.1. Subsequently, in response to Regulatory Guide 1.97 Revision 3, NYPA committed to provide monitors for the main steam lines that met Regulatory Guide 1.97 requirements with a range of 0.04 to 1000 microcuries per cubic centimeters (uCi/cc).

2. **Source**

NYPA responded to NUREG-0737 Item II.F.1 by letter IPN-80-117 dated December 30, 1980, supplemented by letters IPN-81-97 dated December 29, 1981 and IPN-82-33 dated April 20, 1982. NYPA's final response to Regulatory Guide 1.97 was by letter IPN-86-05 dated January 7, 1986. FSAR Section 11.2 states the monitors meet the requirements of Regulatory Guide 1.97 and FSAR Table 11.2-7 sheet 2 of 2 lists a range for channel R-62 of 1E-4 to 1E+1 uCi/cc.

3. **Finding (Discrepancy)**

The actual range of the MSL radiation monitors R-62 (A-D) is not the range committed to in response to NUREG-0737 (i.e., 1E-1 to 1E3 microcuries per cubic centimeters (uCi/cc) based on Xe-133 equivalent), nor the range committed to in the latest response to Regulatory Guide 1.97 Revision 3 [i.e., 4E-2 to 1E3 uCi/cc (based on a radionuclide mix)], nor are the Main Steam Line radiation monitors (R-62) designed to the requirements of Regulatory Guide 1.97 as stated in FSAR Section 11.2. In addition, the range for Channel R-62 listed in FSAR Table 11.2-7 (i.e., 0.0001 to 10 uCi/cc) does not reflect the range capability provided for the monitors to meet the requirements of NUREG-0737.

4. **Status**

Main Steam Line radiation monitors (R-62 A-D) were designed to meet the range requirements of NUREG-0737 Item II.F.1 (i.e., 1E-1 to 1E3 uCi/cc of Xe-133 dose equivalent) and have an installed range capability of 4E-2 to 1E3 uCi/cc of Xe-133 dose equivalent. The as-installed radiation monitors for the Main Steam Lines (R-62 A-D) deviate from the range requirements of Regulatory Guide 1.97 Revision 3, but meet the intent of Regulatory Guide 1.97 and bound all postulated accident activity concentrations.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

C. RANGE OF THE MAIN STEAM LINE RADIATION MONITOR (cont'd)

4. **Status (cont'd)**

The FSAR will be revised in the next scheduled update to clarify Section 11.2.3.1. The revision will clarify that the monitors meet the requirements of NUREG-0737 and that there are three displays for the monitors output, one of which is a local analog indicator with a range of 1E-4 to 1E+1 uCi/cc. FSAR Table 11.2 will be revised to list the range capability provided in response to NUREG-0737. The range listed in FSAR Table 11.2 was a local indicator's range and was the value in question since it does not agree with the Authority's range provided in response to NUREG-0737 or Regulatory Guide 1.97.

Two other indicators, one in the control room and the other local, have the capability to monitor and display the range committed to in the response to NUREG-0737 and Regulatory Guide 1.97 (i.e., 0.04 to 1000 uCi/cc), and meet the intent of the NUREG-0737 range requirement of 0.1 to 1000 uCi/cc of Xe-133 dose equivalent.

5. **Clarification**

NYPA's submittals to the NRC for both NUREG-0737 and Regulatory Guide 1.97, and the NRC's evaluation of those submittals did not note or discuss the basis for the monitor's range. NYPA committed to meet the NUREG-0737 Action Item II.F.1.1 requirement to provide the capability to detect and measure concentrations of noble gas fission products in plant gaseous effluent during and following an accident. The clarification section for NUREG-0737 Action Item II.F.1 noted that the PWR steam safety relief valve discharge required a design maximum range of 1E3 microcurie per cubic centimeter (uCi/cc) of Xe-133. No numerical lower limit was identified except that the range extend from normal conditions which was identified as ALARA. Design range values for monitors employing gamma detectors were to be expressed in Xe-133 equivalent values.

NYPA's commitments for the Regulatory Guide 1.97 Revision 3 range requirements, for the main steam line, was listed as a range of 0.04 to 1000 uCi/cc under Index number 508. NYPA stated that the actual range was in compliance with Regulatory Guide 1.97, but there was no clarification of what the monitor's range was based on, and no reference to the regulatory guide Note 13 concerning range basis. Note 13 of Regulatory Guide 1.97 specified a range of 0.1 to 1000 uCi/cc of an *actual expected radionuclide mix.*

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

C. RANGE OF THE MAIN STEAM LINE RADIATION MONITOR (cont'd)

5. **Clarification (cont'd)**

The range disparity between NUREG-0737 and Regulatory Guide 1.97 is a result of the difference in the source basis specified for the monitor (i.e., Xe-133 versus radionuclide mix). The Main Steam Line (MSL) radiation monitors will provide accurate monitoring of radioactive releases through the main steam lines for the maximum steam line concentrations associated with a postulated design basis Steam Generator Tube Rupture (SGTR) accident. The monitors calculated upper range value is approximately 400 uCi/cc based on a radionuclide mix versus the Regulatory Guide 1.97 limit of 1000 uCi/cc. However, the monitors meet the regulatory criteria since the maximum calculated noble gas concentration in the MSL's for a postulated SGTR accident was determined to range from approximately 8 uCi/cc to 0.2 uCi/cc for a steam line pressure ranging from 755 psig (normal operating level) to 0 psig (following depressurization).

Additionally, the operating total noble gas activity in the primary coolant is 1.36 uCi/cc versus a calculated design basis total noble gas activity of approximately 206 uCi/cc (FSAR Table 9.2-5). These activity concentrations are significantly lower than the monitor's upper range of 400 uCi/cc and well below the regulatory value of 1000 uCi/cc. Therefore, the existing MSL radiation monitors has a range that meets the objective of both NUREG-0737 Item II.F.1 and Regulatory Guide 1.97 and only requires applicable documentation to be clarified and made consistent with the existing design.

D. PROCEDURE REVISIONS FOR OPERATION OF INACCESSIBLE EQUIPMENT POST ACCIDENT

NUREG-0737 Action Item II.B.2 required a design review to determine which types of corrective actions are needed for vital areas and equipment. NYPA was required to provide adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post accident procedural controls. In response to the TMI item, eight manual valves were identified as being required for post accident operation, but located in areas which would be inaccessible because of high dose rates during post accident high head recirculation. Six out of eight valves (in the isolation valve seal water system) were identified as requiring relocation. The remaining two valves of the containment spray system were to be operated in accordance with revised emergency operating procedures, prior to high head recirculation, to meet dose requirements.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

D. PROCEDURE REVISIONS FOR OPERATION OF INACCESSIBLE EQUIPMENT POST ACCIDENT (cont'd)

NYPA stated that it would inform the NRC if the final results of the preliminary analysis alter the committed modifications. However, NYPA did not implement the procedure revisions and notify the NRC.

1. **Commitment**

Preliminary results (shielding analysis to determine the dose rates after a DBA) indicate that there are a total of eight (8) manual valves which require post-accident operation but are located in areas which will be inaccessible during post-accident high-head recirculation. Six of these valves (in the isolation valve seal water system (IVSWS)) will be relocated to a low radiation area. The emergency operating procedures will be revised to ensure that the remaining two valves (in the containment spray system) will be operated prior to the commencement of high-head recirculation phase. The relocation of the IVSWS valves and revision to the emergency procedures are contingent upon the final results of the analysis. NYPA will inform the NRC if the final results alter these modifications.

2. **Source**

NYPA's response to NUREG-0737 Item II.B.2 by letter IPN-83-076 dated September 12, 1983.

3. **Finding (Discrepancy)**

NYPA procedures were not revised to close the two (manual) valves in the containment spray system (i.e., 31/32 containment spray pump discharge isolation valves SI-869A and SI-869B) prior to commencement of high head recirculation nor was the NRC informed that the commitment was changed.

4. **Clarification**

The radiological analysis concerning the accessibility of the containment spray valves (SI-869A and SI-869B) required further assessment to determine the acceptability of manual action without any other changes (e.g., shielding or addition of valve operator). After a LOCA, when the contents of the RWST have been depleted, cold leg recirculation is initiated. The system lineup associated with recirculation is dependent on the size of the break.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

D. PROCEDURE REVISIONS FOR OPERATION OF INACCESSIBLE EQUIPMENT POST ACCIDENT

4. **Clarification (cont'd)**

Indian Point 3 has a unique design which provides for post-LOCA (Large Break) internal recirculation of the post-accident radioactive sump fluids without leaving containment (low head recirculation). Use of the external recirculation system is not expected to begin until 14 hours after the accident.

The time at which valves SI-869A and SI-869B are to be closed would not be during high dose rates because the external recirculation lines would not contain post-accident coolant and therefore be dose limiting. However, for small breaks the RCS pressure is higher than the recirculation pumps head; therefore, they are not sufficient to inject sump water into the RCS under these conditions. Consequently, the system lineup provides for external recirculation after the injection phase as soon as the RWST is depleted, which could occur as soon as two hours after event initiation. Since the containment spray valves can be accessed from both PAB elevation 54'-9" and the 67'-6" elevation, these alternatives were assessed as part of the final dose analysis.

The final dose analysis determined that the dose incurred by accessing/closing the valves would be less than the NUREG-0737 limit of 5 rem, including after external recirculation was initiated. Valve access and closure after a large break LOCA can be accomplished without exceeding the dose limit provided the valves are closed between 2 and 14 hours after the start of the accident.

Valve access and closure after a small break LOCA can be accomplished without exceeding the dose limit even after external recirculation is initiated. The bases for the revised commitment was to allow the operators flexibility to continue containment spray if determined necessary.

Failure to implement the commitment and notify the NRC was determined to be a result of an inadequate commitment tracking system. This inadequacy has been previously identified as restart issue NAP IV.1(NRC-31) and corrective actions described in RCIP Restart Action Plan R-2.1.2.2 and Continuous Improvement Action Plan C-3.1.1.4.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

D. PROCEDURE REVISIONS FOR OPERATION OF INACCESSIBLE EQUIPMENT POST ACCIDENT

5. **Status (Corrective Actions)**

The NRC reviewed RAP Item 2.IV.1 and closed this item in Inspection Report 50-286/94-26. As a result of the final dose assessment, the following administrative controls were revised to reduce overall exposure in line with ALARA considerations and to ensure there is the capability to perform the necessary actions.

- Procedure COL-LV-1, "Locked Valve Check Off List," Revision 26, dated March 29, 1995, was revised to require valves SI-869A and SI-869B to be locked open (LO) at the PAB 67 foot elevation.

- Surveillance test procedure 3PT-M17, "Containment Spray Pump Functional Test," Revision 20, dated May 26, 1995, was revised to specify that isolation valves SI-869A and SI-869B are locked/unlocked and operated during the test using the valves reach rods from PAB elevation 67 ft. The revision's purpose was to specify testing requirements for operation of the valves via the reach rods.

The following procedures contain cautions on high radiation fields and requirements to close the containment spray line isolation valves.

- Emergency procedure ES-1.2, "Post-LOCA Cooldown and Depressurization," Revision 7, dated March 10, 1995, references procedure SOP-CB-11 to isolate lines penetrating containment when equipment is shutdown in post accident conditions (valves SI-869A & B).

- Procedure SOP-CB-11, "Non-Automatic Containment Isolation," Revision 3, dated June 22, 1992, requires closure of valves SI-869A and SI-869B following shutdown of both containment spray pumps and contains precautions and limitations (Section 2) regarding potentially high radiation fields when accessing non-automatic valves post-accident.

COMMITMENT IMPLEMENTATION ASSESSMENT RESULTS

D. PROCEDURE REVISIONS FOR OPERATION OF INACCESSIBLE EQUIPMENT POST ACCIDENT

5. **Status (Corrective Actions) cont'd**

- Emergency procedure ES-1.3, "Transfer to Cold Leg Recirculation," Revision 9, dated May 6, 1995, contains a caution prior to the step for aligning the RHR pump for recirculation that "use of the RHR pumps for recirculation will create extreamly high radiation fields in the pipe penetration area." The procedure contains a step that references procedure SOP-CB-11 to isolate lines penetrating containment when equipment is shutdown in post accident conditions (valves SI-869A & B). A procedure step action requires transfer to emergency procedure ES-1.4, "Transfer to Hot Leg Recirculation," after 14 hours and cautions that High Head Recirculation provides extremely high radiation levels.

LIST OF COMMITMENTS

Number	Commitment	Due
IPN-96-032-01	Revise FSAR Section 6.2.3, External Recirculation Loop Leakage, Item 1 to include the RCP seal return line to the VCT.	Next applicable update (Expected December 1996)
IPN-96-032-02	Revise FSAR Section 11.2 to clarify the fact that monitor R-62 is designed to NUREG-0737 II.F.1, but meets intent of R.G. 1.97; update the range; describe the displays and the basis of the range.	Next applicable update (Expected December 1996)