ATTACHMENT I to JPN-92-060

PROPOSED TECHNICAL SPECIFICATION CHANGE
FLOW REDUCTION OF 10% IN CORE SPRAY PUMP
SURVEILLANCE REQUIREMENTS

(JPTS-89-039)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

3.5 (cont'd)

4.5 (cont'd)

b.	Flow Rate Test - Core spray pumps shall deliver at least 4,265 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure.	Once/3 Months
c.	Pump Operability	Once/month
d.	Motor Operated Valve	Once/month
e.	Core Spray Header Δp Instrumentation Check Calibrate Test	 Once/day Once/3 months Once/3 months
f.	Logic System Functional Test	Once/each operating cycle
g.	Testable Check Valves	Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

Amendment No. ~~40~~, ~~149~~

3.5 (cont'd)

F. ECCS-Cold Condition

1. A minimum of two low pressure Emergency Core Cooling subsystems shall be operable whenever irradiated fuel is in the reactor, the reactor is in the cold condition, and work is being performed with the potential for draining the reactor vessel.

2. A minimum of one low pressure Emergency Core Cooling subsystem shall be operable whenever irradiated fuel is in the reactor, the reactor is in the cold condition, and no work is being performed with the potential for draining the reactor vessel.

3. Emergency Core Cooling subsystems are not required to be operable provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and the water level above the fuel is in accordance with Specification 3.10.C.

4. With the requirements of 3.5.F.1, 3.5.F.2, or 3.5.F.3 not satisfied, suspend core alterations and all operations with the potential for draining the reactor vessel. Restore at least one system to operable status within 4 hours or establish Secondary Containment Integrity within the next 8 hours.

4.5 (cont'd)

F. ECCS-Cold Condition

Surveillance of the low pressure ECCS systems required by 3.5.F.1 and 3.5.F.2 shall be as follows:

1. Perform a flowrate test at least once every 3 months on the required Core Spray pump(s) and/or the RHR pump(s). Each Core Spray pump shall deliver at least 4,265 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure. Each RHR pump shall deliver at least 1,910 gpm against a system head corresponding to a reactor vessel to primary containment differential pressure of $\geq$ 20 psid.

2. Perform a monthly operability test on the required Core Spray and/or LPCI motor operated valves.

3. Once each shift verify the suppression pool water level is greater than or equal to 10.33 ft. whenever the low pressure ECCS subsystems are aligned to the suppression pool.

4. Once each shift verify a minimum of 324 inches of water is available in the Condensate Storage Tanks (CST) whenever the Core Spray System(s) is aligned to the tanks.

4.5 BASES

The testing interval for the Core and Containment Cooling Systems is based on a quantitative reliability analysis, industry practice, judgement, and practicality. The Emergency Core Cooling Systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems will be automatically actuated during a refueling outage. In the case of the Core Spray System, condensate storage tank water will be pumped to the vessel to verify the operability of the core spray header. To increase the availability of the individual components of the Core and Containment Cooling Systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise, the pumps and motor-operated valves are also tested each month to assure their operability. The combination automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling equipment. Consistent with the definition of operable in Section 4.0.C, demonstrate means conduct a test to show; verify means that the associated surveillance activities have been satisfactorily performed within the specified time interval.

The RCIC flow rate is described in the UFSAR. The flow rates to be delivered to the reactor core for HPCI, the LPCI mode of RHR, and CS are based on the SAFER/GESTR LOCA analysis. The flow rates for the LPCI mode of RHR and CS are modified by a 10 percent reduction from the SAFER/GESTR LOCA analysis. The reductions are based on a sensitivity analysis (General Electric MDE-33-0786) performed for the parameters used in the SAFER/GESTR analysis.

The CS surveillance requirement includes an allowance for system leakage in addition to the flow rate required to be delivered to the reactor core. The leak rate from the core spray piping inside the reactor but outside the core shroud is assumed in the UFSAR and includes a known loss of less than 20 gpm from the 1/4 inch diameter vent hole in the core spray T-box connection in each of the loops, and in the B loop, a potential additional loss of less than 40 gpm from a clamshell repair whose structural weld covers only 5/6 of the circumference of the pipe. Both of these identified sources of leakage occur in the space between the reactor vessel wall and the core shroud. Therefore flow lost through these leak sources does not contribute to core cooling.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC Systems are filled provides for a visual observation that water flows from a high point vent. This ensures that

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGE FLOW REDUCTION OF 10% IN CORE SPRAY PUMP SURVEILLANCE REQUIREMENT (JPTS-89-039)

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed change to the James A. FitzPatrick Technical Specifications revises the core spray pump flow rate requirements and the Bases for the core spray pump requirements. The changes to the Technical Specifications are addressed below. Minor changes in format, such as type font, margins or hyphenation, are not described in this submittal. These changes are typographical in nature and do not affect the content of the Technical Specifications.

Page 113, Specification 4.5.A.1.b.

Replace the value "4,625 gpm" with the value "4,265 gpm."

Page 122, Specification 4.5.F.1.

Replace the value "4,625 gpm" with the value "4,265 gpm."

Page 132, Bases 4.5

Add paragraphs four and five that read as follows:

> "The RCIC flow rate is described in the UFSAR. The flow rates to be delivered to the reactor core for HPCI, the LPCI mode of RHR, and CS are based on the SAFER/GESTR LOCA analysis. The flow rates for the LPCI mode of RHR and CS are modified by a 10 percent reduction from the SAFER/GESTR LOCA analysis. The reductions are based on a sensitivity analysis (General Electric MDE-83-0786) performed for the parameters used in the SAFER/GESTR analysis.
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II. PURPOSE OF THE PROPOSED CHANGES

The CS System is one of several Emergency Core Cooling Systems (ECCS) used to mitigate the consequences of loss-of-coolant accidents (LOCAs). Core spray is comprised of two subsystems (independent loops) with each subsystem consisting of a 100 percent capacity motor driven pump, piping, valves and a sparger to transfer water from the suppression pool to the reactor vessel. The A and B core spray lines

enter the reactor vessel through two nozzles located 180° apart to provide physical separation. Each nozzle has a thermal sleeve that is welded into a T box. Two pipes are run from the T box to form a semicircular header with a downcomer at each end. The downcomer has an elbow where the spray lines pass through the upper part of the shroud and into the spray sparger. This configuration is shown on Figures 1 and 2.

The core spray pumps are tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Reference 1) and Technical Specifications 4.5.A.1.b and 4.5.F.1 to ensure that adequate emergency core cooling capacity is available. The current requirement in the Technical Specifications is that core spray pumps deliver at least 4,625 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure. The surveillance test should also account for system leakage that is not delivered to the core. Surveillance testing is conducted in accordance with the In-Service Testing (IST) program.

The purpose of this change request is to reduce the flow requirement for core spray surveillance testing. The reduction in the CS flow requirements is intended to allow the IST program pump performance band to be used to provide the potential for increased system availability. The change will also clarify the testing requirements for system leakage.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

The CS system is an emergency core cooling systems (ECCS) used to mitigate the consequences of loss of coolant accidents (LOCAs) and to provide inventory makeup in the alternate shutdown cooling mode in the event that the suction path from the reactor becomes unavailable for shutdown cooling or reactor inventory is lost. The surveillance testing required by Technical Specifications 4.5 A.1.b and 4.5.F.1 is intended to verify the capability of the core spray pump to deliver to the core the flow assumed in the SAFER/GESTR LOCA analysis (Reference 2). The proposed reduction in core spray pump flow rate will not affect plant safety because the CS system can perform its required functions at the reduced flow rate while accounting for the system leakage. These considerations include:

1. Core Spray Functional Requirement

   A sensitivity analysis (Reference 3) was performed based on the SAFER/GESTR LOCA analysis (Reference 2) to assess the conservatism in current and proposed Technical Specification requirements for ECCS components. The sensitivity analysis varied component performance requirements (e.g., Diesel Generator startup time, pump flow rates, valve stroke times, etc) to determine the sensitivity of the SAFER/GESTR LOCA analysis results for the design basis accident (i.e., recirculation line break). The flow rates for CS, Low Pressure Coolant Injection (LPCI) and High Pressure Coolant Injection (HPCI) were reduced by 10% over their entire range in the analysis. For CS, the reduction was equivalent to a minimum rated flow of 4,163 gpm to the spray nozzles at a reactor vessel pressure equal to 113 psi above containment pressure.

The SAFER/GESTR LOCA analysis supplemented by the sensitivity analysis and an independent safety evaluation (Reference 4) justify operation with the reduced CS pump flow rate. The key issues in this determination were as follows:

- Sensitivity studies performed with the SAFER/GESTR LOCA models demonstrate an increase in fuel peak cladding temperature (PCT) of less than 120°F during the postulated design basis accident due to a reduction in all of the parameters. Since the current limiting licensing PCT is more than 600°F below the 2200°F allowable limit, the reactor core continues to meet the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K with a margin of 500°F. The statistical upper bound PCT remains at least 150°F less than the Appendix K case and will meet the 1,600°F limit of Reference 5.

- Sensitivity studies performed with the SAFER/GESTR LOCA models demonstrate that 88°F of the increase in fuel peak cladding temperature (PCT) is attributable to a 10% reduction of all ECCS flow rates. This leaves a safety margin of more than 500°F.

- An increase in temperature will result in a small increase in the metal water reaction for the limiting break accidents. The results of an earlier LOCA analysis (Reference 6) are more limiting than the results of the sensitivity analysis so the metal water requirements of 10 CFR 50.46 are still met. The containment evaluation of UFSAR Section 14.6.1.3 is also bounding.

- There is no increase in the PCT for the worst case Appendix R fire due to a reduction in core spray flow. The Appendix R analysis (References 7 and 8) assumed operation of one RHR pump in the LPCI mode. The other RHR pumps and the core spray pump were assumed to be inoperable.

- The requirements for inventory makeup to mitigate the consequences of inadvertent draindown while shutdown are bounded by the LOCA. The limiting double ended guillotine break of the recirculation line (4.17 sq. ft.) is larger than any opening associated with draindown. Updated Final Safety Analysis Report (UFSAR) Section 14.6.1.3, indicates that a single core spray system is capable of long term cooling for the LOCA and it is, therefore, adequate for draindown in the cold condition. Since the sensitivity analysis has used the same methodology as the current LOCA analysis, a single CS system at reduced flow is suitable for the cold condition.

2. System Leakage

System leakage is the difference between CS pump flowrate and CS flowrate inside the core shroud. The CS flowrate used in the LOCA analysis (References 2 and 3) is the CS flowrate inside the core shroud.

When the FitzPatrick plant was being designed, leakage was postulated to occur from the thermal sleeve between the T box and vessel nozzle and a quarter inch vent hole in the T box that allowed for release of non condensable gases (Reference 9).

The leakage requirement included in this proposed Technical Specification change is based on an assessment of the actual system leakage (References 9, 10 and 11). The assessment was part of the analysis used to validate CS flowrate after repair of a crack in the core spray piping outside the shroud on the "B" loop. The assessment identifies the elimination of thermal sleeve leakage before plant operation and calculates the upper bound leakage from the upper T box vent hole (0.25 ± .05 inch) as less than 20 gpm. The crack in the "B" loop core spray piping was repaired by welding a clam shell on the upper riser outside the shroud. The weld covers only 5/6 of the circumference of the pipe (attachment 7 to Reference 11) and calculations in Reference 11 conservatively conclude that leakage from the unwelded sector is less than 40 gpm.

Based on the above, the required CS flowrate must allow for a leakage of 20 gpm and 60 gpm to "A" and "B" loops, respectively. Since 4,163 gpm is required for delivery to the core (see item 1), the reduced flow requirement bounds the calculated maximum leak rate.

CS pump surveillance testing to meet Technical Specifications 4.5.A.1.b and 4.5.F.1 and the IST requirement is currently performed in accordance with Surveillance Test Procedure ST-3P (Reference 12). Surveillance Procedure ST-3P is currently adequate to demonstrate the ability of the core spray pump. Allowable ranges for test quantities are specified in accordance with Table IWP-3100-2 to ASME Section XI. Approval of the proposed Technical Specification change will allow ST-3P to be revised for testing at a reduced flow rate.

Operation of the plant in accordance with the proposed amendment will not be a safety concern. The effect of the reduction in CS pump flow is a decrease in the margin between the calculated PCT and the allowable limit. Secondly, there is an increase in margin between the requirements proposed for the Technical Specifications and the ASME Section XI inservice test (IST) reference values. These safety considerations were previously identified in a request to reduce LPCI pump flow by 10% (Reference 13) which was approved as Amendment 171 (Reference 14). The conclusions of the plant's accident analyses as documented in the UFSAR and the NRC staff's SER at operating license stage are not altered by these changes to the Technical Specifications.

The Authority has revised the licensing basis SAFER/GESTR LOCA Analysis (Reference 15) as part of the power uprate evaluation (Reference 16). The SAFER/GESTR LOCA Analysis for power uprate used lower pump flow rates than found in the sensitivity analysis. When approved, the updated analysis will provide the basis for new ECCS flow requirements. Further reductions in flowrate can be requested at that time.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

   The CS system is designed to mitigate the consequences of analyzed accidents and is normally in the standby mode. The proposed changes reduce flowrate which is a reduction in the performance condition required to respond to an accident. This does not effect the manner in which the CS system is tested or its function. Therefore, the changes have no effect on the conditions which could initiate an accident.

   The effect of a 10% reduction in the CS pump flow rate has been analyzed using approved methodology. The PCT increase of 88°F has no significant effect on the existing margin to the 2200°F acceptance criteria. Slight increases in metal water reaction occur. These are not of safety significance because the prior LOCA analyses used a higher PCT and the UFSAR evaluations are based upon metal water reactions due to more severe conditions. There are no changes to the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR). The change, therefore, does not effect continued compliance with 10 CFR 50.46.

2. create the possibility of a new or different kind of accident from any accident previously evaluated.

   The decrease in flowrate for the CS pump is a decrease in the performance requirement for the system. Conditions that could lead to an accident are not changed. There are no changes to the manner in which tests are conducted, no changes to system design and no changes to operating procedures that could result in a new or different kind of accident.

3. involve a significant reduction in a margin of safety.

   The effect of a 10% reduction in the CS pump flow rate has been analyzed using approved methods. Margin of safety is provided by the conservatisms required in Appendix K and by a conservative application of the approved GESTR-LOCA and SAFER Models in NEDE-23785. These margins are not effected by this change.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not adversely affect the ALARA or Fire Protection Programs nor will the changes affect the environment. There will be no plant modifications or testing changes that can have an effect on either the programs or the environment.

VI. CONCLUSION

The changes, as proposed, do not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

1. will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

2. will not increase the possibility of an accident or malfunction of a type different from any previously evaluated in the Safety Analysis Report;

3. will not reduce the margin of safety as defined in the basis for any technical specification; and

The changes involve no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

References relied upon to prepare the Technical Specification change request:

1. ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through Winter 1981 Addenda.

2. NEDC-31317P, "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," dated October 1986 including Errata and Addenda (Proprietary).

3. GE "Sensitivity of the James A. FitzPatrick Nuclear Power Plant Safety Systems Performance to Fundamental System Parameters" dated July 1936 (MDE-83-0786) Proprietary.

4. James A. FitzPatrick Nuclear Power Plant, Nuclear Safety Evaluation JAF-SE-92-146, Rev. 0, "Evaluation of a 10% Reduction In LPCS Pump Surveillance Flow Rate," dated September 1992.

5. NRC letter, C. O. Thomas, to GE dated June 1,1984 regarding acceptance for referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident."

6. GE "Loss-of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Plant, NEDO-21662-2, dated July 1977.

7. GE "Analysis To Extend Operator Action Time For Alternate Shutdown Panels In Support of FitzPatrick Compliance To Appendix R" dated November 1985 (MDE-8137-0585, Revision 2).

8. NYPA letter (JPN-85-090), J. C. Brons to NRC, dated December 17,1985, providing additional information on an exemption request from Section III.L to Appendix R of 10 CFR 50 regarding alternate shutdown capability.

9. GE "Core Spray Line Crack Analysis for James A. FitzPatrick Nuclear Power Plant" dated October 1988 (EAS-64-03243) Proprietary.

10. James A. FitzPatrick nuclear safety evaluation JAF-SE-88-190, "Repair of In-vessel Core Spray Line Using a Welded Clamshell Sleeve," dated October 14, 1988.

11. NYPA letter (JAFP-88-0965), R. J. Converse to NRC, dated October 21,1988, providing additional information on an internal vessel Core Spray System pipe crack.

12. James A. FitzPatrick surveillance procedure ST-3P "Core Spray Flow Rate and Valve Inservice Test," Revision 9, dated December 4,1991.

13. NYPA letter (JPN-90-049), J. C. Brons to NRC, dated June 21, 1990, requesting a change to the Technical Specifications to reduce LPCI pump flow requirements.

14. NRC letter, B. C. McCabe to NYPA, dated July 1, 1991, regarding issuance of amendment 171 to the Technical Specifications.

15. GE NEDC-31317P, "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 1, dated November 1991 (Proprietary).

16. NYPA letter (JPN-92-028), R. E. Beedle to NRC, dated June 12,1992, requesting an Amendment to the Technical Specifications to allow power uprate.

Background documents not specifically referenced.

1. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report Sections 6.3, 7.2, 7.3, 8.3 and Chapters 5 and 14, Revision 5 dated through January 1992.

2. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20,1972, and Supplements.

3. James A. FitzPatrick Nuclear Power Plant, "Inservice Testing Program for Pumps and Valves," dated May 1,1991.

4. James A. FitzPatrick Nondestructive Examination Procedure I.B.I.P. 2, "Visual Examination of the Reactor Vessel and Internals," dated May 14, 1991.

FIGURE 1

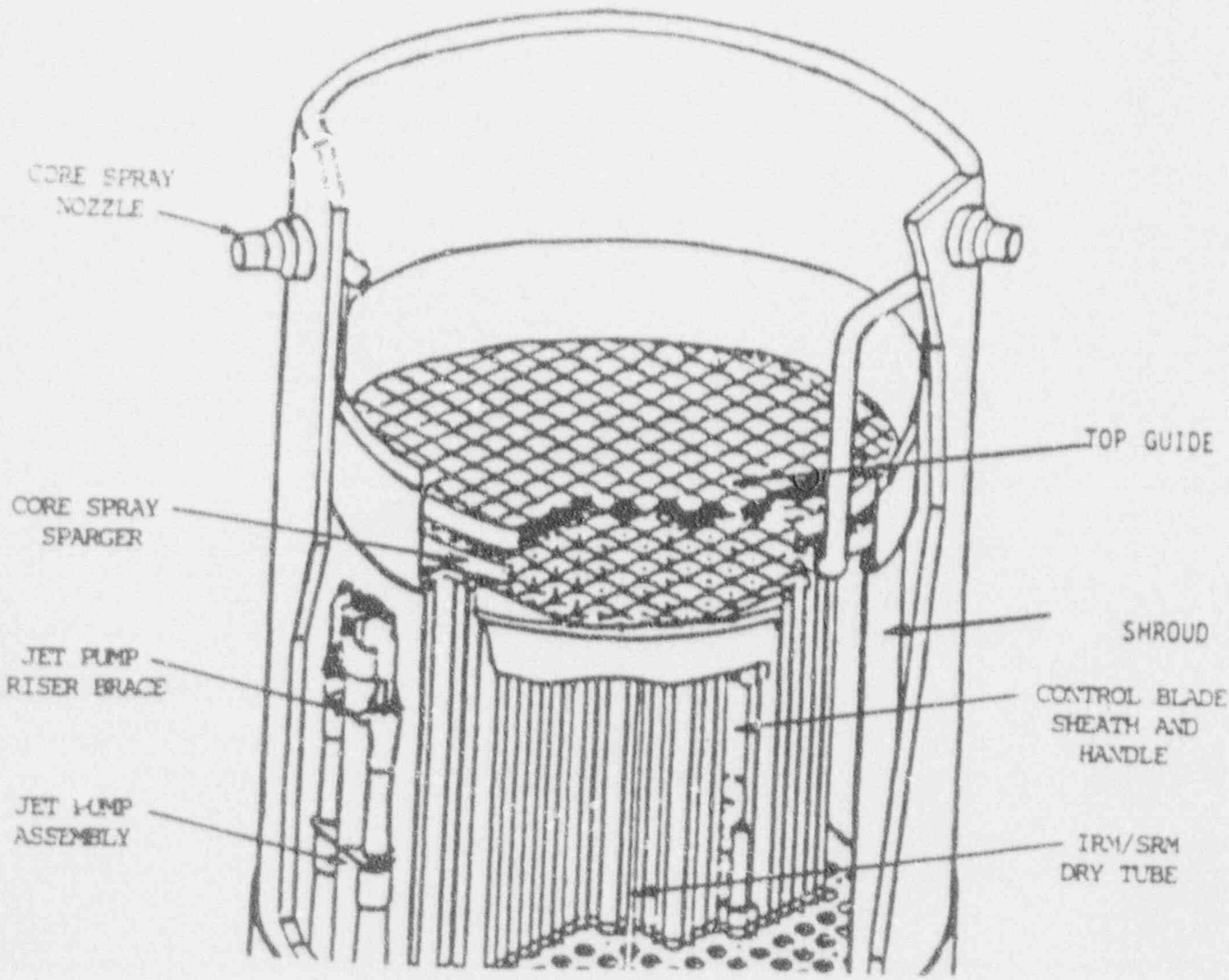
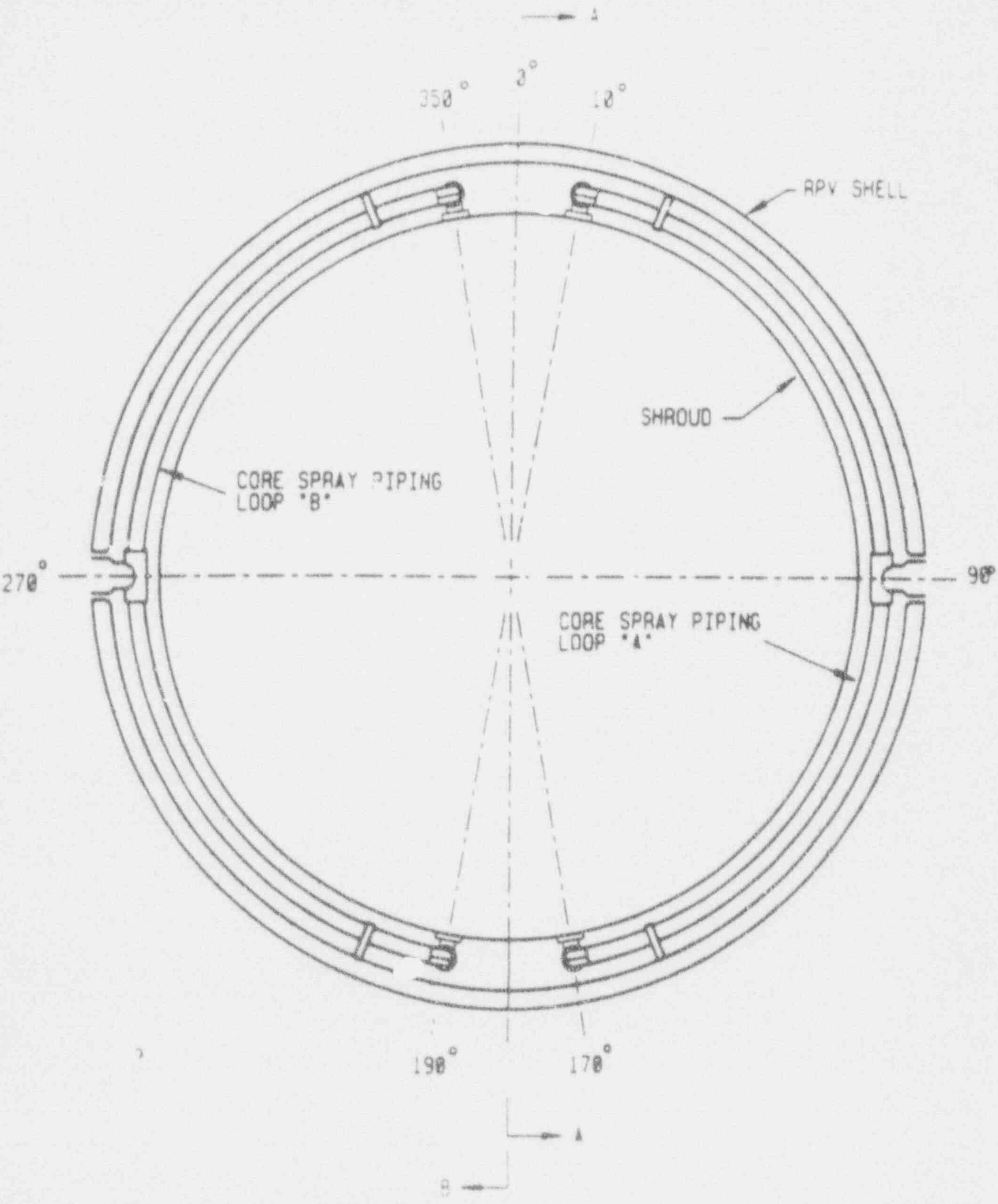

CORE SPRAY NOZZLE
TOP GUIDE
CORE SPRAY SPARGER
SHROUD
JET PUMP RISER BRACE
CONTROL BLADE SHEATH AND HANDLE
JET PUMP ASSEMBLY
IRM/SRM DRY TUBE

FIGURE 2


A
0°
350°
10°
RPV SHELL
SHROUD
CORE SPRAY PIPING
LOOP "B"
270°
90°
CORE SPRAY PIPING
LOOP "A"
190°
170°
A
B

ATTACHMENT III to JPN-92-060

PROPOSED TECHNICAL SPECIFICATION CHANGE
FLOW REDUCTION OF 10% IN CORE SPRAY PUMP
SURVEILLANCE REQUIREMENTS
MARKUP OF TECHNICAL SPECIFICATION PAGES

(JPTS-89-039)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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Insert A

INSERT A

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