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Nuclear Power Plant
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NewYork Power Authority

L. M. Hill
Resident Manager

May 8, 1995
IPN-95-056

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

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Request for Enforcement Discretion for Technical Specification Requirement Pertaining to Residual Heat Removal System

Dear Sir:

The purpose of this letter is to request enforcement discretion for the Indian Point 3 Technical Specification (TS) Section 3.3.A.2 Action Statement which requires that the residual heat removal (RHR) system be restored to operability within 1 hour, or the reactor shall be in cold shutdown condition within the next twenty (20) hours. The enforcement discretion will allow the reactor to remain in hot shutdown with the RHR system inoperable for greater than twenty-one (21) hours, if necessary, to complete a repair to the RHR miniflow instrument line. Enforcement discretion is requested for maximum additional duration of 48 hours to complete the necessary repairs. The justification for this request is provided in Attachment I. Attachment II includes the nuclear safety evaluation.

If this enforcement discretion is granted, it will permit repairs to be completed without going to a cold shutdown or defueled condition, with no detrimental effect on public health and safety.

Subsequent to our verbal request of May 6, 1995, the repair to the line was completed in approximately sixteen (16) hours. Therefore, no enforcement discretion was necessary.

There are no new commitments made in this submittal.

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If you have any questions regarding this submittal, please contact Mr. K. Peters at (914) 736-8029.

Very truly yours,

J. M. Hill
Resident Manager
Indian Point Three Nuclear Power Plant

Attachment: as stated

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ATTACHMENT I TO IPN-95-056

REQUEST FOR ENFORCEMENT DISCRETION FOR RHR PUMP MINIFLOW INSTRUMENT LINE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

1. Requirement Needing Enforcement Discretion

Indian Point 3 Technical Specification 3.3.A.1.c and 3.3.A.2 require:

3.3.A.1 The reactor coolant system T_{avg} shall not exceed 200°F unless the following requirements are met:

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

3.3.A.2 If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within the next 20 hours.

The Authority's understanding of this evolution with PORC's concurrence is that the repair of the RHR piping will commence once the system is declared inoperable, exceed the one hour requirement and enter the 20 hour action statement without proceeding to cold shutdown.

Enforcement discretion is requested which would allow the reactor to remain in the hot shutdown condition with the residual heat removal system inoperable while performing a repair and allow for an additional 48 hours (if necessary) beyond the 20 hour action statement to facilitate the repair. Once the declaration of system operability is performed, IP3 would remain in hot shutdown and exit Technical Specification 3.3.A.2 and continue with the startup.

As noted in our cover letter, the repair was completed and system returned to operable in less than 21 hours, so the Technical Specification was not exceeded, and no discretion was necessary.

2. Description of Circumstances

On May 4, 1995, a deviation event report (DER) 95-1078 was written for a pin hole leak on the flow indicator FI-642 line elbow for the RHR pump miniflow. The reactor is currently in hot shutdown and the Authority is planning to fix the leak in this condition. This enforcement discretion request is necessary to avoid having to bring the reactor to cold shutdown if the repairs cannot be completed within 20 hours. The cause of this leak and any relevant historical events will be researched as part of the DER 95-1078 extent of condition review. As discussed further below, IP3 is in the process of starting up. We need to repair the leak prior to continuing with plant startup, and, as described below, believe there is no safety benefit to going to a lower mode to perform the repair.

3. Safety Significance and Potential Consequences

The Authority is requesting enforcement discretion allowing for continuation of the startup based on criterion 3 of the 10 CFR Part 2 Appendix C guidance in the NRC Inspection Manual, part 9900. Criterion 3 for plants attempting to startup states, "The Technical Specification or other license conditions require a test, inspection, or system realignment that is inappropriate for the particular plant conditions, because it does not provide a safety benefit, or may, in fact, be detrimental to safety in the particular plant condition." The nature of the identified leak is such that a repair can be performed either during the present plant condition of hot shutdown with T_{avg}<350°F or at cold shutdown, or in a defueled condition.

The Authority believes the plant should not be placed in cold shutdown condition in order to repair the piping for the following reasons: When in cold shutdown the RHR system is used for decay heat removal. As per the IP3 Design Basis Document and plant procedures, the RHR miniflow line (valves 743 and 1870) is required to be operable to support RHR pump operation which is consistent with standard pump protection schemes. A repair would require this miniflow line to be taken out of service contrary to the above requirements. Further, during cold shutdown, one or both pumps would be running continuously without miniflow protection. More importantly, when the miniflow line is taken out of service to conduct the repair, a freeze seal will be the only isolation from the reactor coolant system (RCS) since the RHR system is directly connected to the RCS. This situation could lead to an unisolable RCS leak outside containment.

The Authority believes the plant should not be placed in a defueling mode to conduct the repair, although feasible from a RHR system availability standpoint, for the following reason: the hardship and risk associated with defueling the reactor is judged to be greater than conducting the repair in the present plant condition.

The Authority requests discretionary enforcement for the current plant condition (hot shutdown-T_{avg}< 350 °F) to extend the allowed out of service time for the RHR injection train. In the present plant condition one RHR train is required to be operable for injection purposes in the event of a LOCA. Decay heat removal requirements are met using the reactor coolant pumps and the steam generators. In the unlikely event that injection is required during the period when the RHR flow path is isolated for repair of the leak the following is noted: The RHR pumps will be prevented from automatically starting. This is consistent with the Authority's position that manual alignment of the RHR system for injection purposes is feasible in the current plant condition (hot shutdown). The injection evolution can be manually controlled such that injection, only when required, is initiated thereby minimizing the need for the miniflow line. Also considering the present decay heat load, RCS pressure and temperature, the injection of RWST water to the core following a large break LOCA would be adequate with some spillage to the PAB floor (through the unisolated miniflow line under repair). Precautions would be taken to stand ready with mechanical means to limit spillage to the floor if necessary. Additionally, although not required for this mode, the safety injection pumps are available for injection and in

fact are currently operable to provide a flowpath from the RWST to the RCS.

4. No Unresolved Safety Question or Significant Hazards Consideration

The Authority has concluded that this request does not involve a significant hazard consideration in that the request would not:

(i) involve a significant increase in the probability or consequences of an accident previously evaluated. Conducting the repair with the reactor in the hot shutdown condition would not involve an increase in the probability of occurrence nor as discussed previously, the consequences of a design basis accident during the period of this request.

(ii) create the possibility of a new or different kind of accident from those previously evaluated. The proposed request does not involved physical modification to any plant systems or components, only repair. The proposed request does not involve any operations that are different from those described in the FSAR or in plant procedures.

(iii) involve a significant reduction in the margin of safety. Approval of this request involves no reduction in the margin of safety because in the unlikely event that injection is required during the repair period, the injection evolution will be manually controlled and the safety injection pumps would also be operable.

A Nuclear Safety Evaluation (NSE 95-03-170 RHR) was prepared which determined there is no unresolved safety question involved with this repair. (Attachment II)

In summary this request for enforcement discretion will permit temporary relaxation of the Indian Point 3 Technical Specifications which require that above 200°F the RHR system shall be restored to operability in 1 hour or the reactor shall be in cold shutdown condition in 20 hours.

5. Environmental Consequences

No adverse environmental consequences will result from approval of this request. All work will be controlled in an organized manner and means will be available to control any leakage from the 1/2 inch line.

6. Compensatory Actions

The compensatory measures proposed for performing the repair in hot shutdown with T_{avg} < 350°F are as follows:

a. Safety Injection Pumps and SI Accumulators will be operable and aligned to the RCS in the event that injection is required. However, the pumps will be maintained in the trip pullout condition such that operator action will be required to initiate core cooling.

b. The repair work package is written to allow for mechanical means to be available to plug the instrument line if at any time during the repair it is deemed necessary. Communications will also be available between the control room and the job site.

c. The RHR pumps will be placed in trip pull-out to prevent automatic starting if safety injection is required. This measure will allow for manual control of the RHR injection path only when absolutely necessary i.e. large break LOCA has occurred and system pressure is low enough to allow RHR injection.

d. A special evolution briefing will take place prior to job start informing all relevant parties of their role.

7. Justification for Duration of Request

The Authority requests the enforcement discretion for an additional duration of 48 hours to complete the repairs. Although we expect the repair work to be completed within about 14 hours, we are requesting an additional 48 hours as a contingency, in case of problems with establishing welding of the lines and return to service.

8. Plant Operating Review Committee (PORC) Review

The Plant Operating Review Committee has reviewed and approved this request.

9. Satisfaction of Section B of Part 9900: NRC Inspection Manual Guidance Criteria for Plants Attempting to Start Up

The Authority is requesting enforcement discretion allowing for continuation of the startup based on criterion 3 of the 10 CFR Part 2 Appendix C guidance in the NRC Inspection Manual, part 9900. Criterion 3 which falls under the section for plants attempting to startup states, "The Technical Specification or other license conditions require a test, inspection, or system realignment that is inappropriate for the particular plant conditions, because it does not provide a safety benefit, or may, in fact, be detrimental to safety in the particular plant condition." The nature of the identified leak is such that a repair can be performed either during the present plant condition of hot shutdown with T_{avg}<350 $^{\circ}$F, at cold shutdown, or in a defueled

condition. As discussed in Section 3 of this attachment, the safety benefit of transfer to another mode to perform the repair is not justified.

10. Conclusion

As noted previously, we completed this repair and restoration of operability in under 21 hours, and did not need the exercise of enforcement discretion.

ATTACHMENT II TO IPN-95-056

NUCLEAR SAFETY EVALUATION FOR ENFORCEMENT DISCRETION FOR RHR PUMP MINIFLOW INSTRUMENT LINE

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

ATTACHMENT II

New York Power Authority

☑ - IP3
☐ - JAF

10 CFR 50.59 NUCLEAR SAFETY EVALUATION Form

Number: 95-3-170 RHR Revision: 0

Activity: ☐ Modification ☐ Procedure ☐ Test ☐ Experiment ☑ Other

Activity Number: N/A

Title: Repair of Instrument Line for FI-642 with Plant at HSD (TAVG < 350°F)

A. The proposed activity:

1.	☐ does	☑ does not	increase the probability of occurrence of an accident evaluated in the safety analysis report.
2.	☐ does	☑ does not	increase the consequences of an accident evaluated previously in the safety analysis report.
3.	☐ does	☑ does not	increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the safety analysis report.
4.	☐ does	☑ does not	increase the consequence of a malfunction of equipment important to safety evaluated previously in the safety analysis report.
5.	☐ does	☑ does not	create the possibility of an accident of a different type than any evaluated previously in the safety analysis report
6.	☐ does	☑ does not	create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the safety analysis report.
7.	☐ does	☑ does not	reduce the margin of safety as defined in the basis of any Technical Specification.
8.	☐ does	☑ does not	involve an unreviewed safety question based on questions 1 through 7.
9.	☐ does	☑ does not	degrade the Security Plan, Quality Assurance Program or the Fire Protection System.

B. The proposed activity:

1.	☐ does	☑ does not	require a change to the Final Safety Analysis Report as indicated in Section 3 of this Nuclear Safety Evaluation (NSE).
2.	☐ does	☑ does not	require action tracking of the items indicated in Section 5 of this NSE.

C. This proposed activity:

1.	☐ does	☑ does not	require a change to the Technical Specifications.
2.	☐ does	☑ does not	require an Environmental Impact Evaluation.
3.	☐ does	☑ does not	require a change to Design Basis Documents.

Prepared by: Patricia W. Conroy Date: 5-6-95

Reviewed by: [signature] Date: 5/6/95

Recommended: Approval ☒ Disapproval ☐ PORC Mtg. 95-089 Date: 5/6/95

Approved by: [signature] Date: 5/6/95
Resident Mgr. or Designee

Distribution: SRC Coordinator, JAF Site Eng. Mgr/IP3 Technical Services Mgr (annual 50.59 report) RMS-JAF/IP3

NSE 95-3-170 RHR
Rev. 0

1. **PURPOSE AND SCOPE**

A pinhole leak was observed on a 1/2 inch instrument line elbow fitting for FE-642. This safety evaluation considers the different modes of operation for repairing this leak, and outlines the bases for why the leak should be repaired in the current hot shutdown condition, with RCS temperature < 350 degrees F and pressure approximately 1000 psig.

2. **DESCRIPTION OF PROPOSED ACTIVITY**

In order to replace the faulty elbow on the 1/2 inch high side instrument line from the orifice FE-642 to its flow indicator, FI-642, it will be necessary to hold-off portions of the RHR system to isolate all leakage paths to the 1/2 inch hole that will be open during replacement activities.

FE-642 is located in 3 inch RHR miniflow line 337; this line provides pump protection for the RHR pumps in the event these pumps are operated under shutoff head conditions. The purpose of FE-642 is to enable flow measurement through the miniflow line during pump performance testing. The subject faulty elbow is located between the high side tie-in to FE-642 and the root stop valve AC-758A for FI-642. The faulty elbow can be isolated from the pump discharge conditions by closure of containment isolation valves AC-MOV-1870 and AC-MOV-743; no isolation is available from the pump suction conditions without holding off the RHR pumps and associated suction piping.

The following plant conditions have been evaluated for the maintenance repair activities of the faulty elbow: core defuel, cold shutdown (T_{avg}: < 200 degrees F), and hot shutdown (200 degrees F < T_{avg} < 350 degrees F).

With the core defueled, there are no limitations on removing the RHR system from service. During cold shutdown, both trains of RHR are required to be operable and one RHR pump is required to be operating for decay heat removal. During hot shutdown conditions with RCS temperature between 200 degrees F and 350 degrees F, one train of RHR is required to be operable for the ECCS mode of operation; requirements for the RHR system for decay heat removal are dependent upon the status of the RCS components/loops.

The RHR system can be safely taken out of service for an indefinite period following removal of fuel from the reactor vessel. Plant cooldown and core off load would pose a substantial hardship for the plant including occupational exposure to radiation by the plant workers. In addition, the actual process of fuel movement has certain risks associated with it (e.g., fuel handling accident). While these risks are not judged to be significant, they must be weighed relative to the low risk of performing the required maintenance activities during the current plant condition.

Only during cold shutdown must the RHR system be continuously operating for purposes of decay heat removal. Performing the required maintenance activities in cold shutdown would require the RHR mini-flow line to be isolated; in all likelihood this isolation would be provided by closing of the containment isolation valves upstream of FE-642 and applying a freeze seal to the mini-flow line downstream of FE-642. During this condition, the miniflow line would obviously be out of service and yet at least one RHR pump would be required to be operating for decay heat removal. Such a condition is inconsistent with standard pump protection schemes and is contrary to normal plant practice and design basis requirements of having the mini-flow path open prior to starting and during RHR pump operation. More importantly, since the freeze seal represents an isolation boundary from the RCS (since the RHR system is connected to the RCS in this mode of operation), failure of the freeze seal during the maintenance repair activities would initiate a LOCA outside containment. The RHR system would then have to be isolated from the RCS which would then render both Technical Specification trains of RHR decay heat removal inoperable. Performing the maintenance repair activities in cold shutdown is clearly not a reasonable alternative since it is practically judged that failure of the freeze seal is more likely than occurrence of a LOCA in hot shutdown.

Repair of the RHR instrument line would most preferably be accomplished in the current hot shutdown condition, with RCS temperature < 350 degrees F and RCS pressure approximately 1000 psig, for the following reasons:

* The RHR system is not currently in service and decay heat removal is being provided by the steam generators.
* Events requiring the RHR system to perform its ECCS function are highly unlikely in the current plant condition.

* If the RHR system were called upon to mitigate an accident, it could be returned to service in a short period of time.
* Other ECCS subsystems are operable and capable of injecting the contents of the RWST to the core in the extremely unlikely event of an accident.
* The total out of service time of the RHR system is conservatively estimated to be 14 hours which is less than the combined time frames of the 1 hour Tech Spec LCO and 20 hour action statement. This total out of service time includes system hold-off, draining the effected sections of piping, maintenance repair, weld inspection, refill and testing for return to service.

This safety evaluation addresses the above items, and demonstrates that the subject repair activities can be safely performed in the current plant condition.

3. **SAR REVIEW**

The following table summarizes the FSAR and Technical Specification sections which were reviewed, and identifies those sections which will require a revision.

DOCUMENT/SECTION REVIEWED	SECTIONS REQUIRING CHANGES
FSAR SECTION 6.2	NONE
FSAR SECTION 9.3	NONE
FSAR CHAPTER 14	NONE
TECHNICAL SPECIFICATION SECTION 3.1.A & SECTION 3.1.A BASES	NONE
TECHNICAL SPECIFICATION SECTION 3.3.A AND SECTION 3.3 BASES	NONE *

* Technical Specification 3.3.A.2 allows for a 1 hour LCO and 20 hour action statement for an inoperable train of RHR for the ECCS mode of operation. As indicated previously, the total out of service time is conservatively estimated to be 14 hours. It is therefore not expected that a change to this Tech Spec is required. However, as a contingency, a discretionary waiver is being requested from the NRC in conjunction with this NSE to provide for an extension to the 20 hour action limit in the unlikely event the repair activities take longer than expected.

4. REVIEW AND ANALYSIS

During the subject repair, the RHR pumps will be in trip pullout with the suction piping from all three sources isolated (i.e., the RWST, the RCS, and the containment sump). The RHR system will be declared inoperable when the system is isolated in preparation for the maintenance repair activities. The time that the RHR system will remain inoperable has been conservatively estimated to be no greater than 14 hours. Tech Spec section 3.3.A.2 will be in effect, requiring that the system be restored to meet the requirements of 3.3.A.1 (i.e., one RHR pump, heat exchanger, and associated piping and valves operable) within one hour or the reactor shall be in cold shutdown within the next twenty hours. The plant will enter the twenty hour action statement without initiating cooldown operations, as the RHR system operability is expected to be restored within the overall 21 hour time period. If during the course of this repair a major problem arises and it is foreseen that the system cannot be returned to an operable status within the overall 21 hour time frame, the above described enforcement discretion action will be used to allow the plant to remain at hot shutdown until the repair activities have been effectively completed. The likelihood of exceeding the overall 21 hour time frame is extremely small. All parts required for the repair will be pre-fabricated and ready for installation prior to entering the LCO.

The work package will require that a suitable pipe cap or plug be available at the job site as a contingency should Operations require that this line be quickly plugged to support low head safety injection. This work package also requires communications be set up between the job site and the control room prior to start of work. This will ensure contingency actions may be quickly implemented, maintaining this evolution as safe as possible.

The RHR system is required to support core cooling and makeup in the event of an accident, using the normal ECCS alignment. When the RHR system is removed from service to perform the required maintenance activities, these functions can be provided by other means. The RHR system is currently isolated from the RCS, with normal decay heat removal accomplished using the steam generators. The plant can safely stay in the current hot shutdown condition (RCS temperature < 350 degrees and RCS pressure approximately 1000 psig) for an extended period of time without use of the RHR system. Only in the event of an accident would the RHR system be required to provide core cooling and makeup flow. Only a LOCA or MSLB

would result in a rapid decrease in RCS pressure, and only a large break LOCA would result in sufficient depressurization to support low head safety injection flow from the RHR pumps. At this time, there are several methods available to accomplish safety injection.

The RHR system can be returned to service relatively quickly after the system is held off and the instrument line cut. If a LOCA were to occur during the maintenance repair activities, the repair could be completed in a relatively short period of time. If not, the instrument line could be plugged to support low head injection. If the RHR miniflow line remained isolated, the RHR pump(s) could still be operated, however the operators would have to assure that RCS pressure was low enough to support adequate RHR pump flow so as not to dead head the pump(s). Use of the RHR pumps with the miniflow line isolated would thus only be effective in the event of a large break LOCA. If the RHR miniflow line was returned to service, the RHR pumps could be operated regardless of RCS pressure. However, since any such operation would be manually controlled by the operators, in all likelihood the RHR pumps would not be operated if the RCS pressure remained elevated above the shutoff head of the RHR pumps. This is of no consequence since the RHR pumps in such a condition would not be capable of injecting RWST water into the RCS.

If on an emergency basis the repair activities were not completed and instead the line were plugged and the plug subsequently failed with the RHR pump(s) in operation, it is estimated that approximately 65 gpm would be lost out the 1/2 inch line into the PAB assuming a differential pressure of 200 PSID. The actual differential pressure would be much less. However, as indicated above since the RHR pumps would only be used for a large break LOCA with pump flows in the range of 2000 to 3000 gpm, the 65 gpm leak represents a very small percentage of the contents of the RWST being pumped to the RCS. Given a large break LOCA, it is judged that sufficient water from the RWST would be pumped into the RCS and accumulate in the containment sumps to support recirculation pump NPSH requirements.

If a LOCA were to occur during the maintenance repair activities, the contents of the SI Accumulators would feed into the RCS as soon as pressure fell below the accumulator pressure. All four SI accumulators are currently filled, pressurized, and aligned. Although the SI Accumulators are not required to be operable by Tech Specs until RCS temperature is above 350 degrees F, they have been declared

operable in the current plant condition (RCS temperature < 350 degrees F and RCS pressure approximately 1000 psig). This results in an increased margin of safety for the current plant condition.

All three high head safety injection (HHSI) pumps are currently operable (but in the trip pull-out position) and aligned to inject water from the RWST to the RCS. Unlike the RHR pumps, the HHSI pumps can inject water into the RCS at high pressures. Given a large break LOCA, the basis for a minimum safety injection flow rate is to make-up for core boiloff. Since the plant has been shutdown for over two years, the decay heat level is very low. Thus, the required makeup flow rate is very low, certainly far lower than the capacity of the RHR pumps. Emergency Operating Procedure ES-1.3 provides a graph showing the required SI flow rate needed to make-up for core boiloff as a function of time after trip from 100% power. Based on time after shutdown, the SI flow requirement if a LOCA were to occur in this condition is well within the capacity of a single SI pump. As such , for the current plant condition, one SI pump and its associated piping and valves from the RWST to the RCS assures the margin of safety required by the Tech Specs for SI flow to the core.

Given the current plant condition relative to the design conditions of the RCS, 650 degrees F and 2485 psig, a large break LOCA or any size pipe break is extremely unlikely. The RCS is designed for substantially higher stresses than currently exist. More credible conditions of RCS leakage fall within the makeup capacity of the charging pumps, two of which are required to be operable in the current plant condition. The proposed maintenance activities will remove the RHR system from service for up to 14 hours at a time when one RHR pump, heat exchanger, and associated piping and valves are required to be operable for ECCS requirements.

Contingency plans are in place to quickly return the RHR system to service if needed. It is not expected that this repair work will cause the plant to operate outside the limits of the Technical Specifications. While the RHR system is inoperable, other ECCS subsystems, which although not required to be operable, are in fact currently operable in the existing plant condition for core cooling in the extremely unlikely event of an accident. In view of the duration of the repair work, the contingencies in place and the operability of the other SI subsystems, it is concluded that the proposed repair can be safely performed in the current plant condition.

5. **ACTION ITEMS TO BE TRACKED**

NONE

6. **10CFR50.59[b][2] SUMMARY OF ACTIVITY AND NUCLEAR SAFETY EVALUATION**

The proposed maintenance repair activity of the faulty elbow in the high side instrument line associated with FE-642 can be performed safely in accordance with the Technical Specifications with the plant in the current hot shutdown condition (RCS temperature < 350 degrees F and RCS pressure approximately 1000 psig). The RHR system will be inoperable during the conservatively estimated 14 hours required to facilitate the maintenance repairs at a time when one train of the RHR system is required to be operable for ECCS requirements. Contingency plans are in place to quickly return the system to service in the extremely unlikely event of an accident during the brief period of time the system will be out of service. Other ECCS subsystems are operable and can be manually operated to provide for SI flow injection to the core. Based on the extremely unlikely nature of a LOCA occurring in the current plant condition, the relatively brief duration of the repair work, the contingencies in place and the fact that other ECCS subsystems are operable, it is concluded that the proposed repair can be safely performed in the current plant condition.

6.1 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F will not increase the probability of occurrence of an accident evaluated previously in the FSAR. The repair activity is being performed in accordance with the Tech Specs and is incapable of influencing the occurrence of an accident.

6.2 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F will not increase the consequences of an accident evaluated previously in the FSAR. This repair has been evaluated in this safety evaluation to ensure reactor protection is adequate for this mode and that the consequences of postulated accidents for this mode would not increase.

6.3 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F will not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the FSAR. Manual control of the RHR pumps will be initiated only when system pressure is low enough to allow for RHR injection, thereby negating the need for miniflow pump protection and not contributing to an equipment malfunction.

6.4 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F will not increase the consequence of a malfunction of equipment important to safety evaluated previously in the FSAR. The RHR pumps will be manually controlled and only used in the extremely unlikely event of a large break LOCA during the current plant condition.

6.5 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F does not create the possibility of an accident of a different type than any previously evaluated in the FSAR. This repair scenario is performed with the RHR system isolated and if needed the system can be manually aligned with no possibility of creating a different type accident.

6.6 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR. This repair has been evaluated to have no effect on the functioning of equipment needed during the postulated scenarios.

6.7 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F does not reduce the margin of safety as defined in the basis for any Technical Specifications. The proposed repair is expected to be performed within the confines of the applicable Technical Specifications as written or with enforcement discretion.

6.8 Performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F does not degrade the Security Plan, QA Program or the Fire Protection Program. The proposed repair will reinstate the piping to its original design configuration and has no impact on the Security Plan, QA Program or Fire Protection Program.

It is therefore concluded that performing a repair of FI 642 instrument line while in the hot shutdown condition with T_{avg} below 350 degrees F does not involve an unreviewed safety question.

7. REFERENCES

A. WR 95-02206

B. DER 95-1078

C. IP3-DBD-306: SI

D. IP3-DBD-4.2 : RHR

E. DRAWING NO. 9321-F-27353

F. DRAWING NO. 9321-F-27503

G. DRAWING NO. 9321-F-27513, SHEET 1

H. DRAWING NO. 9321-F-27503

I. FSAR SECTIONS 6.2 AND 9.3
FSAR CHAPTER 14

J. TECHNICAL SPECIFICATION SECTIONS 3.1.A and 3.1.A Bases
TECHNICAL SPECIFICATION SECTIONS 3.3.A and 3.3 Bases