Attachment 2 to JAFP-98-0306

JAF-SE-98-013, Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement

New York Power Authority

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

Nuclear Engineering
NUCLEAR SAFETY EVALUATION FORM

☐ IP3 ☒ JAF Nuclear Station   NSE Number: JAF-SE-98-013   Revision #: 1   Full Rev: ☐; Partial Rev. ☒ (See Note 1 below)

Activity Number: F1-97-031   Activity: ☒ Modification ☐ Procedure ☐ Test ☐ Experiment ☐ Other

Title: Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement

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 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. [See Attach. 4.4 Question A7 for guidance on the use of other documents in determining the margin of safety]
8.	☐ does	☒ does not	involve an unreviewed safety question based on questions 1 through 7.
9.	☐ does	☒ does not	degrade the Security Plans (Physical Security Plan, Guard Training & Qualification Plan, Safeguards Contingency Plan), the Quality Assurance Program, the Fire Protection Program, the Environmental Report (including Appendix B to TS, Offdose Calculation Manual, Process Control Manual), or the Emergency Plan.

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.
3.	☐ does	☒ does not	require a change to the item(s) indicated in Section A9 above.

C. This proposed activity:

1.	☐ does	☒ does not	require a change to the Technical Specifications.
2.	☒ does	☐ does not	require a change to Design Basis Documents.
3.	☐ does	☒ does not	require a change to Core Operating Limits Report (COLR), IP3 OS or JAF AP-01.04.

Note 1: Full revisions are complete and do not contain revision bars in the NSE.

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Recommended: Approval ☒ Disapproval ☐; PORC Meeting 98-061 Date: 9/1/98

Approved by: Daniel A. Ruddy / [illegible] Date: 9/1/98
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1.0 PURPOSE

The existing suppression pool suction strainers for Residual Heat Removal and Core Spray system pumps are replaced by plant modification F1-97-031 (Ref. 7.1). The replacement strainers have greater debris loading capacity, to better accommodate the debris generated as a result of a Loss of Coolant Accident, and smaller openings for improved filtration.

Revision 1 has been issued to evaluate a temporary construction opening in the torus, for personnel and equipment access during strainer installation, and the retest developed to verify structural integrity and leak-tightness following reinstatement of the torus opening. Revision 1 also documents completion of Action Items 5.1 and 5.2 by documenting the analytical technique used to evaluate higher accelerations acting on the motor operators for 14MOV-7A/B (Action Item 5.1), and the FSAR change initiated to provide administrative controls on the Core Spray system to bound potential pump degradation from a LOCA downcomer bubble (Action Item 5.2).

2.0 DESCRIPTION OF PROPOSED ACTIVITY

The original suppression pool strainers for the Residual Heat Removal (RHR) and Core Spray (CS) system pumps will be replaced under modification F1-97-031. RHR Pump Suction Strainers 10F-4A & -4B and Core Spray Pump Suction Strainers 14F-2A & -2B will be replaced with larger, high capacity stainless steel strainers with substantially higher debris loading capacity. The existing wire mesh strainers currently in place for RHR and CS, installed over the mitered ends of the torus penetration nozzles for each pump train, will be replaced with a stacked disk design which will connect to new flanges on the penetration nozzles and extend circumferentially into the torus bays. The surface area of the existing and replacement strainers for RHR and CS are given in Table 2.1 for comparison. The design provides high debris loading capacity, low flow entrance velocities and, therefore, low strainer head losses.

TABLE 2.1

STRAINER SURFACE AREA COMPARISON

PUMP	STRAINER ID	PEN NO.	PEN SIZE (in)	SURFACE AREA EXISTING STRAINERS (ft^2)	SURFACE AREA REPELACMENT STRAINERS (ft^2)
RHR 10P-3A, C	10F-4A	X-225A	24	14.9	1128
RHR 10P-3B, D	10F-4B	X-225B	24	14.9	1128
CORE SPRAY 14P-1A	14F-2A	X-227A	16	6.6	336
CORE SPRAY 14P-1B	14F-2B	X-227B	16	6.6	336

Technical Procurement Specification JAF-SPEC-MISC-02871 (Ref. 7.2) specifies the design requirements for the replacement strainer assemblies. Additionally, various analyses/calculations have been prepared to demonstrate the acceptability of the available NPSH, structural integrity of the strainers, supports, penetrations and torus, piping hydraulic flow within the suppression pool, and the heavy load path for installation. (A complete list is included in F1-97-031, Section 10.0) To support installation of the new strainer assemblies, the modification removes the existing suction strainers from the suppression pool, modifies the torus penetration nozzles to accept new flanges, installs new strainer supports on adjacent torus ring girders and modifies supports on the CS suction piping outside containment. No reinforcement of the shell penetrations is required.

Strainer installation will be performed with the torus suppression pool drained in accordance with Technical Specification 3.5.F, which provides ECCS operability requirements with the plant in Cold Condition. The strainers will be lowered into the torus through the Torus Access Hatch Penetration 16X-200A (RS#5). A pre-approved rigging plan will be used to control movement of the strainer assemblies from the Reactor Building Track Bay to the final installed positions in the torus (Ref. 7.3). Load paths and rigging details are defined on Engineering approved sketches. The plan and associated calculations evaluate rigging and safe load paths consistent with criteria in NUREG-0612 (Ref. 7.4).

Post modification examinations will be performed to ensure structural integrity of the containment pressure vessel, torus attached piping, and strainer assembly components affected by this modification meet applicable portions of the ASME code. Preoperational testing will be performed to verify that the suction flow path to each pump is not obstructed and that pump performance satisfies Technical Specification surveillance (Ref. 7.6) and IST acceptance criteria for flow and differential pressure.

A temporary construction opening will be cut in the torus to facilitate personnel and equipment movement during strainer installation. The opening will be made below the water line in the "A" Bay of the torus once the suppression pool is drained and will be repaired following completion of strainer installation. The integrity of the repaired opening will be verified by performance of a Primary Containment Pressurization Test (Ref. 7.39). The test will pressurize the repair to allow visual inspection of the weld for leakage (VT-2), in lieu of the pneumatic test required by ASME Section XI, IWE-5221, and will provide a structural load on the torus chamber to verify integrity has not been diminished by the repair. Request for approval of the alternative test method was submitted to the NRC as Relief Request No. RR-16 (Ref. 7.43).

3.0 SAR REVIEW

3.1 A FOLIO search of the JAF FSAR (Ref. 7.5) and Technical Specifications (Ref. 7.6) was performed using the following words: Residual Heat, RHR, Low Pressure Coolant Injection, LPCI, Emergency Core Cooling, ECCS, Suppression Pool, Loss of Coolant, LOCA, Containment Cooling, Containment Spray, Containment Leakage, Containment Structural Integrity, High Drywell Pressure. A manual search of the FSAR and Technical Specifications tables and figures also was performed. The following FSAR and Technical Specification sections, licensing documents and unincorporated safety evaluations were reviewed for information related to this modification:

- FSAR Sections 1.6.2.11, 1.6.2.12, 4.8, 5.2, 6.4, 6.5, 12.2, 12.5, 14.5, 14.6,16.7, 16.9
- Safety Analysis Report Question and Answer 6.4, Supplement 4 (Ref. 7.7)
- NEDC-32016P-1, Rev.1, Power Uprate Safety Analysis for the James A FitzPatrick Nuclear Power Plant, prepared by the General Electric Company (Ref. 7.8)
- NSE JAF-SE-96-048, Rev. 2, Revision to FSAR to Raise Maximum Allowable Lake Temperature form 82°F to 85°F (Ref. 7.9)
- Technical Specifications 3.1, 3.2, 3.5/4.5, 3.7/4.7, 5.0, 6.20, including the Bases

- Core Operating Limits Report, Rev. 5

3.2 The following sections of the FSAR require revision to incorporate this modification:

- Section 6.6, Incorporation of administrative limits on Core Spray pump operation during normal plant operation. This change documents the requirement to declare a Core Spray pump inoperable prior to operating to ensure the minimum complement of ECCS pumps is available if a LOCA were to occur coincident with pump operation. Analysis postulates Core Spray pump degradation from ingestion of the LOCA downcomer bubble.

- Section 12.5.1.3, Incorporation of changes to the Plant Unique Analysis Report

- Section 12.5.4, The computer code PISTAR was used in the torus attached piping analysis and will be added to this section of the FSAR.

3.3 No changes to the Technical Specifications are required.

4.0 REVIEW AND ANALYSIS

The Residual Heat Removal System (RHR) and Core Spray System (CS) provide cooling of the reactor core under accident conditions to limit fuel clad temperatures. RHR has three modes of post accident operation: Low Pressure Coolant Injection (LPCI), suppression pool cooling and containment cooling. The RHR system in the LPCI mode provides cooling water through the recirculation discharge lines to the reactor core following a Loss of Coolant Accident (LOCA). Once the required coolant level in the vessel is achieved, the RHR system can be realigned to a containment cooling mode. In this mode, the RHR system cools the suppression pool water and can provide containment spray.

In addition to its accident mitigation functions, RHR also removes decay and residual heat from the Reactor Coolant System (RCS) in the shutdown cooling mode, and can supply reactor inventory makeup in the LPCI mode during postulated Appendix R events. RHR pump suction is normally aligned to the suppression pool.

The CS System is provided to protect the core by spraying water over the fuel assemblies to remove decay heat following a postulated design basis LOCA. The protection provided by CS also extends to a small break in which the control rod drive hydraulic pumps, the RCIC System and the HPCI System are all unable to maintain the reactor vessel water level and the Automatic Depressurization System has operated to lower the RCS pressure. The system is normally aligned to take suction from the suppression pool for injection into spray ring headers in the reactor vessel.

4.1 <u>Net Positive Suction Head Evaluation (NPSH)</u>

General Electric Design Specifications 22A1472 (Ref. 7.10) and 22A1435 (Ref. 7.11) required a suction strainer design that would ensure the minimum required NPSH for the RHR and Core Spray pumps could be maintained with their strainer 50% plugged. NPSH for RHR and Core Spray was evaluated in Proto-Power calculation 98-019 (Ref. 7.12) assuming a 50% reduction in strainer surface area to account for debris loading. The most conservative flow path and most limiting NPSH requirement were assumed for each system.. The calculation evaluated NPSH margin at short-term conditions corresponding to maximum pump flow for one train, at the highest suppression pool temperature expected within the first 10 minutes following a design basis LOCA, and at the peak long-term pool temperature assuming the most limiting single failure.

The maximum long-term suppression pool temperature for the current plant licensing basis was calculated in Ref. 7.12 from data documented previously in Safety Evaluations for power uprate and increased ultimate heat sink, NEDC-32016P-1 (Ref. 7.8) and JAF-SE-96-048 (Ref. 7.9), respectively. JAF-SE-96-048 documented the peak suppression pool temperature at uprated power and 87°F lake temperature as 213°F. The NSE stated, however, that the current licensing basis for JAF permits uprated power of 2536 MWt with lake temperature ≤85°F due to limitations in Emergency Service Water and Closed Cooling Water systems. Straight line interpolation between 209°F suppression pool temperature with 82°F lake temperature (the licensing basis condition documented in NEDC-32016P-1 prior to the increased lake temperature design change) and 213°F at 87°F yields 211.4°F for the most limiting long term transient following the design basis LOCA.

The flow rates used in the NPSH analysis for the long-term case reflect the most limiting single failure affecting ECCS pump availability, consistent with Cases C and D in FSAR Section 14.6.1 and the current accident analyses. NPSH margin is affected directly by containment atmospheric pressure, static head, vapor pressure of the suppression pool, losses through the pump suction piping, strainer assemblies, and pump NPSH requirements at the flow being evaluated. It can be assumed that RHR flows higher than the minimum value used in the accident analyses would result in some reduction in pool temperature (with a proportional increase in $NPSH_{AVAILABLE}$). With increasing flow, however, friction losses and pump NPSH requirements increase offsetting, and possibly exceeding, the gain associated with lower pool temperature. Since no correlation has been identified between RHR flow and suppression pool temperature, it can not be stated conclusively that the most limiting case for NPSH margin occurs at the peak suppression pool temperature case evaluated. This potential deficiency has been documented in DER 98-00924.

NPSH analysis for the replacement strainers evaluated short-term operation at 150°F suppression pool temperature consistent with the pool temperature shown in the current GE containment response analysis (Ref. 7.9, 7.13) approximately 600 seconds following a design basis LOCA. Bounding maximum flow rates were evaluated for the strainer being considered. For RHR, bounding flow evaluations were performed assuming operation of one and two pumps per train to conservatively maximize pump NPSH requirements and total flow through the strainers. Flow is actually limited below these values during LPCI operation by pump discharge restriction orifices (Ref. 7.15). Similarly, a bounding flow was used for short-term Core Spray operation, based on the runout flow given in the GE LOCA Analysis performed for power uprate (Ref. 7.8, 7.14). Conditions evaluated were specified in Procurement Specification JAF-SPEC-MISC-02871 (Ref. 7.2) and are summarized in Table 4.1.

TABLE 4.1
NPSH EVALUATION SYSTEM CONFIGURATIONS

SYSTEM	TIME (sec)	NUMBER OF PUMPS PER TRAIN	FLOW (gpm)	SUPPRESSION POOL TEMP (°F)
RHR	600	1	10500	150
	600	2	20800	150
	$6x10^4$	1	7700	211.4
CS	600	1	6000	150
	$6x10^4$	1	4725	211.4

4.2 Containment Overpressure

The NPSH evaluation for RHR and Core Spray pumps (Ref. 7.12) includes use of 2 psig containment overpressure for the most limiting long term transient following the design basis LOCA. The need to credit less than 2 psig containment overpressure for RHR was addressed during the original plant licensing process in the response to Question 6.4 from the AEC (Ref. 7.7) and is reflected in FSAR Figure 6.5-1. The need to credit containment overpressure for long-term Core Spray operation following a design basis LOCA was included in Licensing Amendment No. 239 for power uprate (Ref. 7.16). The approved Technical Specification change revised Technical Specification Bases 3.7 to states that, for the worst case complement of containment cooling pumps (one LPCI pump and two RHR service water pumps) "containment pressure is required to maintain adequate net positive suction head (NPSH) for the core spray and LPCI pumps".

The impact of the power uprate and ultimate heat sink design changes on RHR and Core Spray pump NPSH is discussed in NYPA's response to Generic Letter 97-04 (Ref. 7.17). The letter included containment pressure curves similar to Figure 6.5-1 but updated to reflect the influence of these two design changes on containment response. Revision of Figure 6.5-1 is included in unincorporated NSE JAF-SE-96-048 The curves indicate containment overpressure would be required for adequate RHR NPSH from $3x10^4$ seconds (8.3 hours) until $1.2x10^5$ seconds (33.3 hours) or 25 hours total duration. As was the case on the original curve, the period during which overpressure is required is shown to be coincident with the time at which maximum pressure is postulated in the wetwell and the reduction in NPSH due to temperature is offset by increased containment pressure.

Replacement of the suppression pool suction strainers for RHR and Core Spray will not impact the time and duration for which overpressure is required as compared to the current licensing basis. As shown in the NPSH analysis (Ref. 7.12), the head loss across the RHR and Core Spray suction strainers assuming 50% of the surface area is blocked would be less than 1 ft at long-term design flow rates. ECCS NPSH calculations performed since initial plant design have assumed head loss values ranging from 0 to 1 ft (original plant calculations contained no vendor data or references for strainer head loss). Losses calculated for the replacements, therefore, are consistent with the existing analyses.

4.3 Strainer Hole Sizing

General Electric Design Specifications 22A1472 (Ref. 7.10) and 22A1435 (Ref. 7.11) required a suction strainer design that would adequately filter out particles of sufficient size to prevent clogging spray nozzles or cyclone separators in pump seal flush piping. General Electric Service Information Letter No. 323 (Ref. 7.18) later recommended that ECCS pump suction strainer mesh sizes be checked to ensure particulates passing through the strainers could not plug orifices associated with the cyclone separators. The construction of the replacement stacked disc strainer assemblies uses perforated plate with a $^3/_{32}$" hole diameters assembled on central core tubes. The strainer openings were specified to ensure removal of particles less than the diameter of the most limiting orifice associated with either pump system.

4.4 Air / Steam Ingestion

Strainer layout and proximity to the SRV T-Quenchers and LOCA downcomers were determined from a three-dimensional AutoCAD™ model of the torus and the strainer layout drawings. The size and location of the bubbles discharged from the SRV T-Quenchers and the LOCA downcomers were determined and the potential for ingestion into strainers evaluated in DE&S calculation A384.F02-04 (Ref. 7.19). The possibility for pump degradation or condensation induced water hammer were evaluated in accordance with NUREG-0897 (Ref. 7.20).

The initial bubble from the LOCA downcomers (predominately nitrogen) is discharged within a fraction of a second. If RHR or Core Spray pumps are not in operation, any air that might be forced into the suction strainer during the growth or detachment of the bubble would rise and escape from the strainer before the pumps are started. If an RHR pump were operating when a LOCA occurred, the evaluation in A384.F02-04 (Ref. 7.19) determined that the resulting air/nitrogen bubble should pass through the system before the pump is required to perform its ECCS function. Using the guidance provided in NUREG-0897, the evaluation concluded that the impact on pump performance should be negligible.

The LOCA downcomer air/nitrogen bubble could have a more significant impact on the Core Spray pumps if a LOCA occurred while a pump is in operation. Core Spray surveillance testing currently is performed with that pump declared inoperable. Reactor operation with one Core Spray pump inoperable is permissible, under the limiting conditions of operation in Technical Specification 3.5.A, as long as all other active components of the Core Spray and LPCI systems are operable. As discussed in the Bases, the minimum complement of operable subsystems is maintained under this LCO such that no single failure of ECCS equipment during a LOCA would result in inadequate cooling of the reactor core. Since the conditions of the LCO would be applicable while a Core Spray pump is operating, potential degradation of that pump by a LOCA bubble would be bounded by the existing evaluation. Revision of the applicable surveillance and operating procedures for Core Spray pump operation is included in F1-97-031 to add requirements regarding LCO entry during planned operation of a Core Spray pump. This requirement is documented in a revision to FSAR Section 6.6 included in F1-97-031, Rev.0.

Following Safety Relief Valve (SRV) actuation, the compressed air/nitrogen in the discharge line is discharged within a fraction of a second into the suppression pool forming a high pressure bubble. Depending upon their proximity, there is a potential for the air bubble to overlap ECCS suction strainers. The possibility of air ingestion into a Core Spray pump is not considered credible because the strainers are not located in the vicinity of the T-Quenchers. A384.F02-04 (Ref. 7.19) determined that air ingestion into the RHR system would be possible if a pump were in operation during SRV discharge. The impact on performance would be negligible since air ingestion would last only a fraction of a second and the ingested air would be mixed with water.

4.5 <u>Vortex Limits</u>

The minimum required water level above the strainers to prevent vortexing was calculated in DE&S calculation A384.F02-03 (Ref. 7.21) using results from test data and analysis conducted at Alden Research Laboratory (ARL). The results were compared against EPRI test data for a prototype PCI strainer. The analysis demonstrates that as a result of the relatively low entrance velocities associated with the PCI Sure-Flow™ strainer core tube design and large surface areas specified for the RHR and Core Spray strainers, only partial submergence of the strainer modules is required to prevent vortex formation. For the bounding flow rate of 20800 gpm for one RHR strainer and maximum 6000 gpm for Core Spray, the calculation concludes that vortexing is precluded with 1 ft of water above the strainer centerline. The minimum submergence to prevent vortexing is specified in EOP-2 (Ref. 7.22) and EOP-3 (Ref. 7.23) Vortex Limits curves for the range of RHR and Core Spray pump flow rates. Existing EOP calculations and the EOP Vortex Limit curves will be revised to incorporate minimum submergence data for the new strainers.

Revision of the EOP calculation and curves is tracked in F1-97-031.

4.6 NUREG-0783 Suppression Pool Temperature Analysis

Following Safety Relief Valve (SRV) actuation, long-term steam blowdown raises pool temperature. As the pool temperature increases, the condensation rate at the T-Quenchers is reduced. Adequate subcooling is required to ensure steam bubbles formed by SRV discharge are prevented from being ingested into the ECCS suction strainers.

The impact of increased ultimate heat sink temperature and power uprate operating conditions on a long-term SRV discharge event was evaluated in GE analysis GE-NE-T23-000737-01 (Ref. 7.13). The evaluation determined that the peak local suppression pool temperature with 87°F RHR Service Water temperature satisfies the NUREG-0783 requirement to have at least 20°F subcooling at the SRV quenchers (Ref. 7.24). The requirement in NUREG-0783 is intended to prevent unstable condensation at the SRV discharge into the suppression pool. Replacement of the RHR and Core Spray suction strainers will not affect either the bulk or local suppression pool temperature, nor the assumptions used in the analysis. The results of the previous analysis, therefore, remain valid.

4.7 Structural Analysis and Qualification

The original RHR and Core Spray torus suction inlets had small cantilevered strainers attached to the penetration nozzles. Since the installation of the new strainers involves adding piping and components inside the torus, new hydrodynamic load generation and reanalysis were required. The new load generation and piping system analysis followed the existing methodologies documented in the PUAR to the extent practicable. Techniques used which differ from the original plant unique analysis are summarized in a supplement to the FitzPatrick Plant Unique Analysis Report (PUAR) (Ref. 7.35). The revised PUAR has been approved and accepted, and can be issued following approval of this NSE.

The replacement strainers and strainer supports have been qualified for the loads, load combinations and acceptance criteria established under the Mark I Containment Reevaluation Program. Evaluation of Mark I hydrodynamic loads postulated for the replacement RHR and Core Spray suction strainers followed the generic requirements of NUREG-0661 (Ref. 7.25) and the plant unique methodologies as specified in the PUAR (Ref. 7.26, 7.27). In addition to deadweight, thermal and seismic loads, the strainers, strainer supports and torus attached piping were evaluated for Condensation Oscillation and Chugging including Fluid Structure Interaction effects, Safety Relief Valve (SRV) air bubble and water jet loads, LOCA water jet and air bubble loads, and Pool Swell fall back loads. A complete listing of the structural analyses is included in F1-97-031.

As required by NUREG-0661 and as documented in the PUAR, the Torus Attached Piping (TAP) was evaluated using the design rules of the ASME Boiler and Pressure Vessel Code, Subsection NC, 1977 Edition with Addenda up to and including Summer 1977 Addenda.

4.8 Hydrodynamic Loads

The development of submerged structure loads for the RHR and Core Spray strainers followed the requirements of the Load Definition Report Application Guides (LDR) (Ref. 7.28), NUREG-0661 (Ref. 7.25) and the PUAR (Ref. 7.26, 7.27) except for the use of reduced acceleration drag volumes to account for holes in the stacked disk strainer assemblies. The original plant unique analysis and Mark I LDR Application Guides do not provide guidance for modeling perforated structures.

In the determination of the acceleration drag volumes for the hydrodynamic loads acting on the replacement strainer assemblies, a hydrodynamic mass coefficient, C_m, was determined for the strainers taking into account their geometry and perforated characteristics. The theoretical acceleration drag volume for a cylinder with no holes is 2.0 times the displaced water volume (i.e., C_m=2.00). Testing of prototypical strainers was performed to measure C_m coefficients for the PCI Sure-Flow™ stacked disk strainer design (Ref. 7.29, 7.30). The tested strainer, although basically cylindrical in shape, had holes that allow water to pass thereby reducing the effective added mass. Tests were performed on a strainer comparing the same shape with and without holes. The tests concluded that, for a strainer with perforated plate with 1/8" holes and 40% open area, a C_m coefficient of 50% of the coefficient for the same strainer without holes (i.e., C_m=1.00) could be justified and shown to be conservative.

The C_m coefficients obtained from the testing of the prototype strainer were adjusted to account for differences between the prototype design and the FitzPatrick-specific design. The geometry of the FitzPatrick strainers, including stacked disk width and gap width, is consistent with the tested strainers except that the FitzPatrick strainers utilize perforated plate with 3/32" holes and 33% open area. The effect of the reduced hole size and flow area will tend to increase the differential pressure across the perforated plate, thereby potentially increasing the C_m coefficient. This effect was conservatively estimated to be 75% of the coefficient that would be applied to the same strainer without holes (i.e., C_m=1.50). The methodology used to determine C_m assures no reduction in the safety margin for the FitzPatrick containment and ECCS design basis.

Subsequent to the development of the C_m coefficient used in the qualification of the FitzPatrick strainers, additional testing was performed on several Sure-Flow™ strainers (Ref. 7.31). These strainers had varying design parameters, including perforated plate hole size, disk and gap widths, core tube diameters, etc. These tests confirmed that the C_m value used in the analysis is conservative. Empirical equations derived from the test results predicted that C_m would be no more than 0.71 for these geometries. Supplemental analysis of the Core Spray suction piping was performed using a C_m = 0.75 to reduce excess conservatism and more realistically determine the accelerations acting on Core Spray isolation valves, 14MOV-7A/B.

Analysis of fluid structure interaction (FSI) loads, documented in the existing PUAR, utilized Continuum Dynamics, Inc. (CDI) FSI software and FitzPatrick PUAR torus shell accelerations. FSI loads for the new strainers were developed using FitzPatrick PUAR torus shell accelerations and bounding attenuation curves developed by NUTECH during the Mark I Program. These changes to the analytical technique are documented in the PUAR revision (Ref. 7.35)

4.9 Load Combinations

In the structural analysis for the new strainer assemblies, independent dynamic loads were combined using the Square Root Sum of the Squares (SRSS) method. This method was used in the evaluation of certain RHR and Core Spray components in the original Mark I analysis. The use of the SRSS method, however, was not documented in the PUAR and subsequent SER which state that the Mark I dynamic piping loads were combined by absolute sum. Combining piping system responses from independent dynamic loads by the SRSS method has been shown in NEDE-24632 (Ref. 7.32) to meet the requirements of Paragraph 4.4.3 of NUREG-0661. NUREG-0661 states that, as an alternative to absolute sum combinations, the cumulative distribution function method (CDF) may be used on a component specific basis to combine independent dynamic loads and the CDF combined stress values must show a nonexceedance probability of 84%. NEDE-24632 used the CDF methods to show that the SRSS combination of independent Mark I dynamic loads has a nonexceedance probability of at least 84%. Based on the review of NEDE-24632, the NRC has accepted the use of the SRSS combination of piping system responses due to independent dynamic loads, as documented in a 1983 letter from the NRC to General Electric (Ref. 7.33). This method has been widely used throughout the industry on Mark I plants. A supplement to the PUAR has been prepared, which documents the use of SRSS for FitzPatrick (Ref. 7.35).

4.10 Integral Welded Attachments

Paragraph NC-3645 of ASME III requires that attachments to Class 2 piping shall be designed so as not to cause flattening of the pipe, excessive localized bending stresses or harmful thermal gradients in the pipe wall. The code does not provide a methodology or criteria for evaluating the effects of attachments to piping. Furthermore, the Mark I Plant Unique Analysis Application Guide (Ref. 7.34), the FitzPatrick PUAR and its subsequent SER, and NUREG-0661 also do not describe a methodology or criteria used to evaluate local attachments to piping. The ASME Code, in general, does not limit the engineer's/designer's choice of the methods used to meet the code rules. Therefore, to meet the requirements of NC-3645, Code Cases N-318 and N-392 were chosen as reasonable methods for the qualification of local welded attachments on Class 2 piping. Code Cases N-318 and N-392 were specifically chosen for the following reasons: they are of the same vintage as the design Code (1977 edition with addenda through Summer 77) and their methodology has been accepted previously by the NRC. As the Code Cases are being used solely as a method to meet the requirements of the Code of Record, no code reconciliation is required. This use of Code Cases N-318 and N-392 to evaluate the effects of attachments to Class 2 piping has been documented in a supplement to the PUAR (Ref. 7.35).

4.11 Ring Girder Supports

Each RHR strainer assembly will be supported by three ring girder supports. A single ring girder support will be required for each Core Spray assembly. During accident conditions, the ring girders are loaded by loads on the shell including thermal and seismic, direct submerged structure loads and/or pool loads, and reaction loads of attached equipment and supports.

The suppression chamber structure (torus shell) and ring girders were evaluated with the additional loads resulting from the strainer assemblies. The load definitions and combinations provided in NUREG-0661 were followed for performing the analysis. Existing ring girder stresses, taken from the Mark I Program ring girder evaluations, were added to the resulting stresses due to piping reaction. In addition, the effects of these additional loads on the ring girder to torus shell weldments and on the shell were evaluated.

Based on the evaluation of the torus shell/ring girder structure, it was determined that the center ring girder supports for the RHR strainer assemblies would require the addition of stiffeners at the ring girder web/flange and flange cover plates to reinforce the ring girders. With these modifications, the analysis demonstrated that the stresses on the torus shell, ring girders and component structures meet the allowable stress limits of ASME Section III as originally evaluated in the PUAR and as required by NUREG-0661.

4.12 RHR Bellows Assemblies

The new design for the RHR suction strianers utilizes expansion joints (bellows) to connect the strainer assemblies to the torus penetrations. The expansion joints isolate and decouple the torus attached piping and minimize the impact of the new strainers on the penetrations. Analysis has demonstrated that piping stresses and penetration loads are within their applicable ASME code limits, and existing analyses for RHR pump suction piping external to the torus remain valid.

Qualification of the bellows assemblies used displacements for different load events obtained from the piping analysis. The displacements were used to calculate the limiting displacement combinations and associated numbers of cycles. Stainless steel bellows assemblies were specified with stainless steel covers capable of withstanding a transverse pressure of 10 psi, due to hydrodynamic loads, and accelerations of 15g.

4.13 Core Spray Pump Suction Valves

The accelerations acting on Core Spray pump suction isolation valves, 14MOV-7A/B, were predicted to increase significantly as a result of the strainer replacement (Ref. 7.40, 7.41). The valves have been evaluated to ensure structural integrity would be maintained under all design conditions (Ref. 7.36). The predicted accelerations acting on the motor operators are bounded by the Limitorque qualification (Ref. 7.42)

4.14 Suppression Pool Inventory

The replacement strainers and ring girder supports will displace approximately 300 ft^3 more water volume than the existing components. Technical Specification 3.7.1, Containment Design, Primary Containment, limits suppression pool level to between 13.88 and 14.00 ft. The Bases relate the specification requirement to the downcomer submergence levels assumed in the containment analyses. The water level height of the suppression pool is necessary to assure complete condensation of steam during the blowdown phase of an accident. The Technical Specification limitations will remain unchanged and, therefore, the modification will have no impact on downcomer submergence.

As discussed in Technical Specification Bases 3.7, the minimum downcomer submergence results in a suppression pool water volume of approximately 105,900 ft^3. The suppression pool water provides the heat sink for the RCS energy release following a postulated LOCA. It must be able to "absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig". The additional volume displaced by the new suction strainer assemblies was addressed by JAF-CALC-MISC-02924 (Ref. 7.37). The analysis, which calculated the actual water volume of the 16-sided torus and included the submerged structures, concluded that the actual suppression pool inventory exceeds the Technical Specification value even with the larger strainers installed.

4.15 Temporary Torus Construction Opening

A temporary construction opening will be cut in the torus to facilitate personnel and equipment movement during strainer installation. The opening will be made in the "A" Bay of the torus in approximately the same location as a previous opening used during the Mark I Containment modifications. The opening will be repaired using the same or equivalent material and weld processes qualified to meet notch toughness requirements of the original construction code. Weld integrity will be verified by volumetric, surface and visual examinations of 100% of the final weld and adjacent weld area, and surface examination of the weld preparations on the shell and hatch pieces. Structural and leakage testing will be performed at the calculated peak drywell pressure.

A qualitative evaluation concluded that the temporary opening will not adversely impact the structural capacity of the torus during the construction period. The evaluation, documented in ECN F1-97-031-001, considered the relatively small size of the opening, the static loadings applicable while primary containment is out of service, the proximity to an adjacent ring girder, and the structural support provided by the remaining steel plate. Qualification and NDE of the repair weld ensure the strength of the welded joint will be equal to or greater than that of the base material. Thus the repair will not affect primary containment structural or pressure boundary integrity.

4.16 Primary Containment Pressurization Test

The integrity of the repaired torus opening will be verified by performance of a Primary Containment Pressurization Test once the suppression pool has been refilled (Ref. 7.39). The test will pressurize the primary containment volume to an internal pressure ≥ 45 psig for a minimum of one hour. The reinstated torus opening, therefore, will be subjected to a hydrostatic pressure >45 psig. Visual inspection of the weld for leak-tightness (VT-2) will be performed under the hydrostatic conditions in lieu of the pneumatic test required by ASME Section XI, IWE-5221. Additionally, the test will provide a load on the torus chamber to allow visual verification of structural integrity at the repaired area. The test pressure is based on the 45 psig peak primary containment internal pressure postulated for the design basis loss of coolant accident (Ref. 7.6).

The procedure developed to pressurize primary containment is similar to the leakage rate surveillance testing described in the JAFNPP Primary Containment Leakage Rate Testing Program as referenced in the FSAR. The prerequisites, precautions and limitations, and steps used to establish primary containment test pressure contained in the current integrated leak rate test (ILRT) procedure have been incorporated into the Primary Containment Pressurization Test. Since this post modification test is limited to verification of local leak-tightness and structural integrity in the area of the repair weld, it does not include requirements for pressure maintenance/stabilization beyond the nominal one hour period and does not measure pneumatic leakage through containment penetrations. Plant configuration and conditions established for the post modification test are consistent with those in the periodic leakage rate surveillance. The one time test will have no impact on the leakage rate surveillance program described in the FSAR and does not alter the description of any test or procedure in the safety analysis report.

Testing will be performed with the plant in Cold Shutdown in accordance with Technical Specification 3.7. The Specification requires primary containment integrity only when the reactor is critical or RCS temperature is above 212°F. Thus primary containment will not be required during the testing period.

4.17 10CFR 50.59 Evaluation

Replacement of the suppression pool suction strainers for the RHR and Core Spray pumps, and cutting a temporary construction opening in the torus:

1. **does not increase the probability of occurrence of an accident evaluated in the safety analysis report.**

   RHR and Core Spray are accident mitigation systems. The replacement of the suppression pool suction strainers does not alter any initial condition or assumption used in the accident analysis. FSAR Section 12.2A.6 identifies four major accidents in the Chapter 14 analysis. These events are:

   - Control rod drop accident

- Postulated piping breaks inside containment
- Postulated piping breaks outside containment
- Refueling accident (fuel handling accident)

Replacement of suction strainers in the suppression pool has no impact on control rod or fuel handling accidents. The passive pump suction strainers are not pressure retaining components and their failure can not affect RCS pressure boundary integrity.

FSAR Section 12.2A.7 describes other design basis events including events in which inadvertent operation of ECCS pumps result in a decrease in moderator temperature, and the loss of RHR-shutdown cooling during the low pressure portion of a normal reactor shutdown and cooldown. Since this modification involves only the installation of passive pump suction strainers, there is no impact on any ECCS control function and this modification can not increase the probability of an accidental pump start. The transient in which an RHR Shutdown Cooling suction line becomes inoperative resulting in an increase in reactor vessel water temperature during normal reactor shutdown and cooldown, and during refueling condition, is described in FSAR Section 14.5.8.1. This modification does not alter the overall configuration of any ECCS pump suction line and adds no components that could affect shutdown cooling. The function of the replacement pump suction strainers in the suppression pool will be identical to the original components. Installation of strainers with significantly larger surface area and higher debris capacity will reduce the probability of a loss of pump NPSH and cannot cause an accident.

The replacement of the suppression pool strainers for these systems with the new strainer assemblies, therefore, will not increase the probability of an accident previously evaluated in the SAR.

2. **does not increase the consequences of an accident evaluated previously in the safety analysis report.**

The current accident analyses verify that one RHR pump and one Core Spray pump can provide adequate core cooling following a design basis LOCA. Installation of new passive suction strainers will not affect the operation or performance of the RHR and Core Spray systems. No new failure mechanisms are introduced which could affect the availability of an RHR and Core Spray pump or heat exchanger.

Analysis performed for the RHR and Core Spray system pumps verified that, with a 50% reduction in suction strainer surface area, adequate Net Positive Suction Head (NPSH) would be available under limiting accident conditions assumed in the current accident analysis. The replacement strainers have substantially larger surface areas than those currently installed, increasing the likelihood of long-term pump availability following a LOCA.

Thermal input to the suppression pool is not increased and the heat capacity of the RHR heat exchangers is not changed. Although the replacement strainers will displace more water volume than the existing strainers, the suppression pool level specified in Technical Specification 3.7 and minimum pool water volume described in Technical Specification Bases 3.7 are met. The water volume assumed available as the heat sink for RCS energy release, following a postulated LOCA, and downcomer submergence levels required to provide complete condensation of steam discharged to the pool through the downcomers, therefore, will be unaffected.

The strainers, strainer supports, torus structure, and torus attached piping have been analyzed and qualified for the loads, load combinations and acceptance criteria established under the Mark I Containment Reevaluation Program. The design ensures the strainer assemblies can withstand seismic events and hydrodynamic loads associated with a design basis LOCA without loss of structural integrity. Air and steam bubble formation associated with LOCA and SRV discharges have been evaluated to ensure pump operation will not be adversely affected by the new strainer configurations.

The installation of substantially larger suppression pool strainers for RHR and Core Spray system pumps will not adversely impact the ability of either system to perform its accident mitigation functions during postulated LOCA's. Analysis has demonstrated that the modification will not adversely affect containment pressure vessel integrity or alter the heat removal functions of the suppression pool. This modification, therefore, will not increase the current predicted radiological release for a design basis accident and there will be no increase in the consequences of an accident evaluated previously in the SAR.

Further, no change implemented by this modification affects the ability of RHR to provide makeup water during a postulated Appendix R event or to provide shutdown cooling.

3. does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the safety analysis report.

The surface areas of the new strainer assemblies are significantly greater than those of the current strainers to accommodate a larger accumulation of debris, while still meeting the net positive suction head requirements of the systems' pumps. Their design minimizes entrance velocities and, therefore, head loss, and precludes the formation of a vortex from entraining air into the system. The NPSH analysis (Ref. 7.12) verifies that, with 50% of their suction strainer surface blocked by debris, the RHR and Core Spray pumps would have adequate $NPSH_{AVAILABLE}$ to perform their accident mitigation functions in accordance with the original licensing basis. The perforated holes in the disk assemblies were sized to ensure particles do not pass through which could plug system spray nozzles or orifices.

The head loss across the new strainer assemblies is shown to be less than 1 ft with a 50% reduction in surface area, consistent with assumptions used in original plant NPSH analyses. Since the modification does not affect thermal input to or heat removal capacity from the suppression pool, $NPSH_{AVAILABLE}$ is not reduced from that in the existing licensing basis. The magnitude and duration of containment overpressure required to ensure adequate NPSH for RHR and Core Spray pumps is unchanged.

Air and steam bubble formation in the suppression pool as a result of SRV or LOCA downcomer discharge has been evaluated (Ref. 7.19). The analysis demonstrated that there would be no impact on the accident mitigation performance of either an RHR or Core Spray pump if a LOCA occurred while the pumps were in stand-by. The LOCA bubble would discharge and escape the suppression pool before pump start. If a pump were in operation when the LOCA occurred, only the Core Spray pumps would be susceptible to degradation as a result of the air/nitrogen bubble cleared from the downcomers. Following a LOCA with one train of Core Spray in operation, a worst case LOCA bubble in the torus could engulf the Core Spray strainer resulting in excess of 15% void fraction of the suction piping fluid. The potential impact would be failure of the pump to develop flow as required by the accident analysis.

The Core Spray pumps normally are maintained in stand-by for automatic start in response to LOCA conditions. Core Spray surveillance testing currently is performed with the pump in test declared inoperable. Reactor operation with one Core Spray pump inoperable is permissible, under the limiting conditions of operation in Technical Specification 3.5.A, as long as all other active components of the Core Spray and LPCI systems are operable. In this condition, no credit is taken for this pump in the accident analysis. Since the pump is already assumed to be unavailable to perform its accident mitigation function, the postulated degradation of the pump by LOCA downcomer air/nitrogen ingestion does not increase the probability of a previously evaluated malfunction. Additional administrative controls will be put in place to ensure that the system is declared inoperable whenever a Core Spray pump is operated and a LOCA is credible.

The new strainer assemblies were structurally evaluated to ensure integrity with debris loading in excess of 50%. The replacement ECCS suction strainers, torus attached piping, torus penetrations and supporting ring girders were evaluated for dead weight, pressure, thermal and seismic loads, safety relief valve discharge loads, and Mark I containment hydrodynamic loads acting directly on the submerged assemblies and indirectly by torus motions transmitted to the piping at the torus attachment points. The design ensures the integrity of the strainers and associated piping and supports under design basis conditions and verifies that the additional loads on the primary containment pressure vessel remain within code allowables.

The new passive suction strainer assemblies are similar to the existing strainers. The design of the new strainer assemblies has been fully evaluated through analyses, calculations, and scaled testing to ensure the design requirements of seismic, structural loading, hydrodynamic loading, and NPSH are met. This modification does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the safety analysis report.

Temporary removal and reinstallation of a section of the pressure vessel for personnel and equipment access during the construction period will not adversely affect primary containment. The removal, repair and post modification testing will be performed with the plant in Cold Shutdown in accordance with Technical Specification 3.7. The Specification requires primary containment integrity only when the reactor is critical or RCS temperature is above 212°F. Thus primary containment will not be required during the construction and testing period.

Qualification and NDE of the repair weld ensure the strength of the welded joint will be equal to or greater than that of the base material. Post modification testing (Ref. 7.39) will verify leak-tightness and structural integrity of the repaired area. Testing for the repaired torus opening will be performed at the peak primary containment internal pressure for the design basis loss of coolant accident. The 45 psig nominal test pressure is below the 56 psig design of the vessel (Ref. 7.5). The test, therefore, will provide assurance that the pressure vessel can perform is safety function under deign basis accident conditions while remaining well within the design limits of the vessel. Activities associated with the temporary opening, therefore, will not increase the probability of primary containment failure under design basis conditions.

4. does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the safety analysis report.

The modification does not introduce any new failure modes to any existing equipment. For the most limiting case evaluated in the current GE LOCA analyses (Ref. 7.13, 7.14), operation of only one RHR (LPCI) pump and one Core Spray pump is assumed for long-term LOCA response. This is consistent with the worst case containment response configuration described in FSAR Table 14.6-1 and is based on failure of the power supplies to the safety trains. The new strainers are passive components and, therefore, have no active failure modes which could affect the availability of the safety systems.

The modification has been evaluated to ensure adequate NPSH would be available for long-term operation, that air or steam ingestion would not degrade pump performance, and that the heat sink functions of the suppression pool are not adversely affected. Structural analyses have demonstrated the structural integrity of the strainer assemblies under hydrodynamic loads associated with a design basis LOCA and that the installation will not adversely affect the containment pressure vessel.

Air and steam bubble formation in the suppression pool as a result of SRV or LOCA downcomer discharge has been evaluated (Ref. 7.19). The analysis demonstrated that there would be no impact on the accident mitigation performance of either an RHR or Core Spray pump if a LOCA occurred while the pumps were in stand-by. The LOCA bubble would discharge and escape the suppression pool before pump start. If a pump were in operation when the LOCA occurred, only the Core Spray pumps would be susceptible to degradation as a result of the air/nitrogen bubble cleared from the downcomers. Following a LOCA with one train of Core Spray in operation, a worst case LOCA bubble in the torus could engulf the Core Spray strainer resulting in excess of 15% void fraction of the suction piping fluid. The potential impact would be failure of the pump to develop flow as required by the accident analysis.

The Core Spray pumps normally are maintained in stand-by for automatic start in response to LOCA conditions. Core Spray surveillance testing currently is performed with the pump in test declared inoperable. Reactor operation with one Core Spray pump inoperable is permissible, under the limiting conditions of operation in Technical Specification 3.5.A, as long as all other active components of the Core Spray and LPCI systems are operable. In this condition, no credit is taken for this pump in the accident analysis. Since the pump is already assumed to be unavailable to perform its accident mitigation function, the postulated degradation of the pump by LOCA downcomer air/nitrogen ingestion does not increase the consequences of a previously evaluated malfunction. Additional administrative controls will be put in place to ensure that the system is declared inoperable whenever a Core Spray pump is operated and a LOCA is credible.

The installation of larger suction strainers does not alter the assumptions made in the safety analysis, does not degrade the performance or impact the independence of any safety significant system. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR, therefore, will not increase as a result of this modification.

5. **does not create the possibility of an accident of a different type than any evaluated previously in the safety analysis report.**

The new strainers are passive, non pressure-retaining components which perform the same function as the existing strainers. No new operating/failure modes are introduced by the replacement and no new system interactions are created. The evaluations performed verified the modification will not adversely affect the safety functions of the suppression pool or the structural integrity of the primary containment pressure boundary. No other safety systems are affected by this modification. This modification, therefore, does not create the possibility of an accident of a different type.

6. **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.**

The new strainers are passive, non pressure-retaining components as are the existing strainers. No new failure modes or system interactions are introduced by the modification and independence of the safety trains is not affected.

The RHR and Core Spray pumps provide reactor core and containment cooling under accident conditions. The failure of an RHR or Core Spray pump to provide flow as a result of air/nitrogen or steam ingestion has been evaluated. The analysis demonstrated that there would be no impact on the accident mitigation performance of either an RHR or Core Spray pump if a LOCA occurred while the pumps were in stand-by. The LOCA bubble would discharge and escape the suppression pool before pump start. If a pump were in operation when the LOCA occurred, only the Core Spray pumps would be susceptible to degradation as a result of the air/nitrogen bubble cleared from the downcomers. Following a LOCA with one train of Core Spray in operation, a worst case LOCA bubble in the torus could engulf the Core Spray strainer resulting in excess of 15% void fraction of the suction piping fluid. The potential impact would be failure of the pump to develop flow as required by the accident analysis. Since, this is an existing type of failure, no new failure mode is created by air/nitrogen ingestion.

The Core Spray piping system has been evaluated also to ensure structural integrity would be maintained. The fluid in the suction piping is not pressurized and the gas bubbles would make up a bubbly flow mixture. Thus, for the suction piping, the gas bubbles would not collapse or expand, and little if any pressure perturbation to the piping system would result. Gas bubbles reaching the pump would not collapse but would be compressed to the discharge pressure. The bubbles would then become entrained in the discharge flow without the pressure perturbation associated with the collapse of vapor bubbles. The gas bubbles entrained in the flow would be discharged out of the system. Thus, the system would experience little if any pressure perturbation and pressure boundary integrity would not be unacceptably challenged.

The materials of the strainer assemblies are fully compatible with the Reactor Coolant System and support the structural requirements of the design. The new strainer assemblies and supports have been analyzed for design basis seismic and Mark I hydrodynamic loads. The analyses also confirm that the installation will not adversely affect the integrity of the containment pressure vessel. The strainers will be mounted to prevent any seismic/blowdown event from inducing a failure of the strainer assemblies, suction piping or the penetration. System response is not altered due to the new strainers.

This modification, therefore, does not create the possibility of any malfunctions of equipment important to safety of a different type than previously evaluated in the SAR.

7. **does not reduce the margin of safety as defined in the basis for any Technical Specification.**

The Bases for Technical Specification 3.5.A describe the combinations of operable RHR and Core Spray subsystems specified by the Limiting Conditions of Operation as those needed to assure the availability of the minimum reactor cooling during a LOCA. These ensure that no single failure of ECCS equipment occurring during a LOCA will result in inadequate cooling capacity. Installation of larger, passive suction strainers with higher debris capacity will not alter the operation or performance of the RHR or Core Spray systems. The minimum number and/or combinations of pumps required by the existing Technical Specifications will not be affected nor will the impact of any single failure.

Technical Specification 3.7.1 limitations on suppression pool level will remain unchanged. The water level required to ensure the minimum downcomer submergence required for complete condensation of steam during the blowdown phase of an accident will be maintained.

Analysis has shown that the 105,900 ft^3 suppression pool water volume currently specified in Technical Specification Bases 3.7 will be met even with the installation of larger RHR and

Core Spray suction strainers (Ref. 7.37) The suppression pool water volume required to act as the heat sink for the RCS energy release following a postulated LOCA will not be reduced below that discussed in the Bases.

The design of the new strainer assemblies has been fully evaluated through analyses, calculations, and scaled testing to ensure the design requirements of seismic, structural loading, hydrodynamic loading, and NPSH/head loss are met. The systems will continue to operate as currently designed and the margins of safety defined for the Technical Specification will not be reduced.

Post modification testing for the temporary torus opening (Ref. 7.39) will be performed with the plant in Cold Shutdown in accordance with Technical Specification 3.7. The Bases for this Specification state that the integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the RCS integrity exists, i.e. whenever the reactor is critical and above atmospheric pressure. Operability of primary containment, therefore, will not be required while the test is in progress.

The test procedure disables high drywell pressure Reactor Protection System (RPS) scram and ECCS initiation signals, and the high drywell pressure permissive to Emergency Diesel Generator (EDG) vital load sequencing. As discussed in the Bases to Technical Specification 3.1, instrumentation for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the ECCS initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent return to criticality.

Technical Specification 3.1 requires the high drywell pressure scram function to be operable only for Startup and Run modes. The trip function is not required when the reactor is subcritical with RCS temperature less than 212°F or when primary containment integrity is not required. Since testing will be performed with the plant in Cold Condition, the drywell high pressure scarm function is not required to be operable.

Technical Specification 3.2 requires drywell high pressure ECCS initiation to be operable whenever any ECCS subsystem is required by Specification 3.5. Technical Specification 3.5.F requires a minimum of one low pressure Emergency Core Cooling subsystem to be operable whenever fuel is in the reactor, the reactor is in Cold Condition, and no work is being performed with the potential for draining the reactor vessel. The Specification stipulates that Secondary Containment Integrity shall be established if the ECCS operability requirements can not be satisfied. The test procedure includes steps to ensure suppression pool level remains within Technical Specification 3.7 limits and a RHR loop is maintained in standby. Since the requirement for drywell high pressure initiation can not be met, the test procedure requires the establishment of Secondary Containment Integrity prior to disabling drywell high pressure ECCS initiation.

The Bases for 3.5.F.2 state that one LPCI subsystem, consisting of one motor-driven pump, associated piping, and valves, can provide sufficient vessel flooding capability to recover from an inadvertent vessel drain-down when the reactor is in a Cold Condition. The procedural requirement to maintain one RHR loop in standby and suppression pool level within Technical Specification limits ensures the availability of one LPCI subsystem.

Drywell high pressure is described as a permissive for EDG vital load sequencing in Technical Specification 3.2, not identified as required for EDG operability. Table 3.2-2 states that the

4kV Emergency Bus Undervoltage Timers initiate sequential starting of vital loads in conjunction with low-low-low reactor water level or high drywell pressure. Disabling the high drywell pressure signal to the EDG sequencer will not affect the reactor water level signal and will not affect their availability.

Performance of the primary containment pressurization test (Ref.7.39), therefore, will not reduce the margin of safety as defined in the basis for any Technical Specification.

8. **does not involve an unreviewed safety question based on questions 1 through 7.**

9. **does not degrade the Security Plans (Physical Security Plan, Guard Training & Qualification Plan, Safeguards Contingency Plan), the Quality Assurance Program, the Fire Protection Program, the Environmental Report (including Appendix B to TS, Offdose Calculation Manual, Process Control Manual), or the Emergency Plan.**

No protected or vital area barriers are affected, no security equipment is altered or affected in any way, nor is the configuration of any equipment altered which could interfere with the operation of any security equipment.

All components affected by the proposed modification will be procured and installed in accordance with the applicable requirements for Category I components. There will be no impact on the Quality Assurance Program.

The proposed changes will not impact the site fire protection program. No fire barrier is affected, no flammable or combustible material is added or removed from any area of the plant, and no fire detection/suppression equipment or emergency lighting is impacted. FPES-04A Exhibit 1, (Ref. 7.38) has been prepared, reviewed and approved. The modification has been evaluated for impact on the Fire Protection Program and Appendix R Safe Shutdown Analysis and the commitments made therein are not impacted, invalidated, or affected.

Touch-up painting, required at the completion of strainer and support installation, will be controlled by existing plant procedures which specify use of pre-approved paints. The modification has no other effect on air or water quality, does not involve use of any hazardous substance, does not affect any property outside the plant buildings, does not introduce any new effluent paths, involve storage of any radioactive material, or affect the Meteorological Tower.

The installation of replacement strainers for the RHR and Core Spray pumps will not affect the operation or performance of any component or system required for accident mitigation. EOP revisions will be required to incorporate new NPSH and vortex limit curves. Revision of these curves does not alter any of the steps prescribed by the EOP's. This modification has no impact on Emergency Plan staffing, notification, monitoring or reporting systems. No structures outside the reactor building will be affected.

5.0 ACTION ITEMS TO BE TRACKED

5.1 Completed (See JAF-SE-98-013, Rev. 0)

5.2 Completed (See JAF-SE-98-013, Rev. 0)

5.3 Primary containment can not be declared operable following reinstallation of the temporary torus opening until NRC approval of Relief Request No. RR-16 (Ref. 7.43) is received. (ACT-98-35307)

6.0 10CFR50.59(B)(2) SUMMARY OF ACTIVITY AND NUCLEAR SAFETY EVALUATION

Modification F1-97-031, Rev. 0, "Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement", installs new suppression pool suction strainers for Residual Heat Removal and Core Spray system pumps. The replacement strainers are substantially larger than the current strainers and provide additional margin for debris loading following a design basis accident. Consistent with the original licensing basis for FitzPatrick, the replacement strainers have been evaluated to ensure adequate NPSH would be available to the pumps, under the most limiting conditions analyzed in the current LOCA analysis, with the suction strainers 50% blocked. The strainer assemblies and supports have been analyzed and qualified for seismic and Mark I hydrodynamic loads specified in the FSAR and Plant Unique Analysis Report.

A temporary construction opening will be made in the torus to facilitate personnel and equipment movement for the strainer installation. Integrity of the repaired opening will be ensured by qualification and non-destructive examination of the weld, and by post modification testing. A primary containment pressurization test will be performed at peak accident pressure to verify leak-tightness at the weld and structural integrity of the repaired area.

A review of the modification in accordance with 10CFR 50.59 concluded that the modification does not increase the probability or consequences of an accident or of a malfunction of equipment important to safety previously evaluated in the safety analysis report. Further, the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created. The margin of safety as defined in the Technical Specification Bases is not reduced and no Technical Specification change is required. This change, therefore, does not involve an unreviewed safety question.

7.0 REFERENCES

7.1 JAF Modification F1-97-031, Rev. 0, Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement

7.2 Specification No.: JAF-SPEC-MISC-02871, Rev. 5, Technical Procurement Specification for Residual Heat Removal (RHR), Core Spray (CS), High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Suction Strainers

7.3 JAF-RPT-MISC-02940, Rev. 0, Rigging Plan - ECCS Strainer Replacements

7.4 NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, July 1980

7.5 FSAR, 1997 FSAR Update

7.6 Technical Specifications 3.1, 3.5/4.5, 3.7/4.7, 6.20 Updated through Amendment 240

7.7 James A. FitzPatrick Safety Analysis Report Question and Answer 6.4, Supplement 4

7.8 NEDC-32016P, Rev. 1, Power Uprate Safety Analysis for the James A. FitzPatrick Nuclear Power Plant, prepared by GE Nuclear Energy

7.9 JAF-SE-96-048, Rev. 2, Revision to FSAR to Raise Maximum Allowable Lake Temperature from 82°F to 85°F, Modification F1-97-016

7.10 General Electric Design Specification 22A1472, Rev. 1, Residual Heat Removal System (with Steam Condensing)

7.11 General Electric Design Specification 22A1435, Rev. 1, Core Spray System

7.12 Proto-Power Corp. Calculation No. 98-019, Rev. C, Emergency Core Cooling and Reactor Core Isolation System Pump Suppression Pool NPSH

7.13 GE-NE-T23-0737-01, James A FitzPatrick Nuclear Power Plant Higher RHR Service Water Temperature Analysis, General Electric Company, August 1996

7.14 NEDC-31317P, Rev. 2, James A FitzPatrick Nuclear Power Plant Safer/Gestr-LOCA Loss-of-Coolant Accident Analysis, GE Nuclear Energy, April 1993

7.15 JAF Modification M1-91-242, Rev. 0

7.16 Ralph E. Beedle, New York Power Authority, letter to U.S. Nuclear Regulatory Commission, JPN-92-028, Proposed Changes to the Technical Specifications Regarding Power Uprate (JPTS-91-025), 6/5/92

7.17 J. Knubel, New York Power Authority, letter to U.S. Nuclear Regulatory Commission, JPN-97-039, 90-day Response to NRC Generic Letter 97-04 Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps, 12/23/97

7.18 General Electric Service Information Letter (SIL) No. 323, Suppression Pool Suction Strainer Mesh Size Mismatch with Emergency Core Cooling System (ECCS) Pump Seal Orifices, 1980

7.19 DE&S Calculation, A384.F02-04, Rev. 1, ECCS Suction Strainer Bubble Ingestion

7.20 NUREG-0897, Rev. 1, Containment Emergency Sump Performance

7.21 Duke Engineering & Services Calculation No. A384.F02-03, Rev. 0, RHR, CS, HPCI and RCIC Suction Strainer Vortex/Minimum Submergence

7.22 Emergency Operating Procedure EOP-2, Rev. 4, RPV Control

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