Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315 342 3840

December 11, 2000
JAFP-00-0288

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Completion of Actions for NRC Bulletin 96-03, Potential Plugging of ECCS Suction Strainers

References:

1. NYPA Letter, M. J. Colomb to the NRC, "Information Regarding NRC Bulletin 96-03, Potential Plugging of ECCS Suction Strainers," (JAFP-98-0306), dated September 16, 1998

2. NYPA Letter, M. J. Colomb to the NRC, Change in Commitment Made in Response to NRC Bulletin 96-03, Potential Plugging of ECCS Suction Strainers," (JAFP-99-0284), dated October 22, 1999

3. NRC letter to the New York Power Authority, "Meeting Summary of May 21, 1998, Regarding Emergency Core Cooling Suction Strainer Modifications (TAC No. M96146)," dated June 4, 1998

4. NYPA Letter, M. J. Colomb to the NRC, "Response to NRC Bulletin 96-03," (JAFP-96-0439), dated October 29, 1996

5. NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," dated May 6, 1996

Dear Sir:

This letter reports completion of actions required to satisfy the concerns of NRC Bulletin (NRCB) 96-03 (Reference 5) for the James A. FitzPatrick Nuclear Power Plant (JAFNPP).

IE73

Passive strainers were installed in the fall of 1998 during RFO 13. A final report was deferred, however (see Reference 1), due to a decision (involving significant person-rem savings) to leave a small quantity of microporous insulation in the drywell pending receipt of test results associated with that type of insulation. The JAFNPP committed to providing a final report summarizing actions taken relative to NRCB 96-03, within 90 days of acceptance of a test report and subsequent calculation showing that the microporous insulation could remain in place, or within 30 days after startup from RFO 14 if the test report and calculation showed that additional insulation required replacement. This commitment also retained the conservative licensing basis assumption of 50% strainer blockage, until final actions were completed.

The aforementioned test results on microporous insulation indicated that removal of additional insulation was required. Accordingly, additional microporous insulation was removed during RFO 14, assuring acceptable performance of the ECCS suction strainers for satisfying the criteria of NRCB 96-03.

Strainer inspections were performed during RFO 14. These inspections demonstrated satisfactory suction strainer performance.

In addition to the foregoing, the JAFNPP committed to desludging the suppression pool and to inspecting ECCS suction strainers during RFO 14. The commitment for desludging the suppression pool was later changed to once every other RFO, commencing with RFO 15, based upon engineering calculations and observed sludge generation rate (see Reference 2).

These actions complete JAFNPP requirements regarding NRCB 96-03.

The modifications referred to in this letter were performed under the provisions of 10CFR50.59. Copies of the Nuclear Safety Evaluations for these modifications were previously provided by attachment to Reference 1. Updated revisions of these Nuclear Safety Evaluations are included as Attachments 1 and 2 of this letter. The attached evaluations include a summary description of both the modifications and the design considerations involved. Additional information, including detailed design calculations and test results, is available for inspection at the JAFNPP.

There are no commitments contained in this letter. If you have any questions, please contact Mr. John Hoddy at (315) 349-6538.

Very truly yours,

Michael J. Colomb
Site Executive Officer

STATE OF NEW YORK
COUNTY OF OSWEGO
Subscribed and sworn to before me this 11th day of Dec. 2000.

ANNA F SWAIN
Notary Public - State of New York
~~NO. 01SW5054555~~
Qualified In Oswego County
My Commission Expires Jan 16. 2002

Anna F. Swain

MJC:JRH:las

Cc: next page

Cc: Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

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

Mr. Guy Vissing, Project Manager
Project Directorate I
Division of Licensing Project Management
U.S. Nuclear Regulatory Commission
Mail Stop 8C2
Washington, DC 20555

Attachment 1 to JAFP-00-0288

Nuclear Safety Evaluation JAF-SE-98-013

Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement

Nuclear Engineering
NUCLEAR SAFETY EVALUATION FORM

☐IP3 ☒ JAF Nuclear Station NSE Number: JAF-SE-98-013 Revision #: 3 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 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. *[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.

Prepared by: A.L. Krinzman *A L Krinzman* Date: 7/26/00

Reviewed by: T.M. Driscoll *Thomas M. Driscoll* Date: 7/31/00

Recommended: Approval ☑; AZ Disapproval ☐; PORC Meeting 00-053 Date: 8/15/00

Approved by: *[signature]* Date: 8/15/00
Site Executive Officer or Designee

Distribution: SRC Coordinator, JAF Dir Design Eng/IP3 Systems Engineering Mgr (annual 50.59 report) RMS-JAF/IP3 Preparer

1.0 PURPOSE

The original suppression pool suction strainers for Residual Heat Removal and Core Spray system pumps were replaced by plant modification F1-97-031 during Refuel Outage (RO) 13 (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 was 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 documented 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).

Revision 2 was issued to include discussion of the ECCS suppression pool suction strainer hole size relative to fuel bundle Lower Tie Plate flow holes and the impact of the strainers on local area dose rates. Revision 2 also updated the evaluation to reflect changes in ring girder reinforcements and documented completion of Action Item 5.3 by documenting NRC acceptance (Ref. 7.54) of Relief Request No. RR-16 for alternative testing of the temporary torus opening.

Revision 3 addresses changes to the plant licensing basis for the ECCS suction strainers to incorporate criteria outlined in NRC Bulletin 96-03 (Ref. 7.2) and the revised peak suppression pool temperature derived in GE-NE-T23-00766-00-01 (Ref. 7.56).

2.0 DESCRIPTION OF PROPOSED ACTIVITY

2.1 NRC Bulletin 96-03

NRC Bulletin 96-03 (Ref. 7.2) notified licensees of the operating events and detailed analysis, documented in NUREG/CR-6224 (Ref. 7.3), which demonstrated the potential for common-cause failure due to excessive buildup of debris from thermal insulation, corrosion products and other particulates on ECCS pump strainers. The NRC staff concluded that this issue must be resolved to ensure compliance with 10 CFR 50.46 for long-term core cooling capability following a LOCA. Noting that Regulatory Guide 1.82, Revision 2 (Ref. 7.4) provided an acceptable method of ensuring compliance with 10 CFR 50.46, all BWR licensees were requested to implement "appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA".

The Bulletin requested implementation of corrective actions by the end of the first refueling outage starting after January 1, 1997. The NRC staff identified three potential resolution options in the Bulletin but allowed alternative solutions which could be demonstrated to provide an equivalent level of assurance that the ECCS will be able to perform its safety function following a LOCA. Consequently, the BWR Owners' Group developed a Utility Resolution Guidance (URG) document (Ref. 7.5) to provide detailed guidance on performance of plant-specific analyses consistent with Reg. Guide 1.82, Revision 2.

JAF selected implementation of a design change, designated as Option 1 in the Bulletin, by which high capacity, passive strainers would be installed in the suppression pool for each ECCS pump. Design considerations cited in the Bulletin for this option include providing a strainer design with sufficient capacity to ensure that debris loadings for the most severe postulated LOCA do not cause a loss of NPSH for the ECCS and a design which ensures structural integrity for the strainers under LOCA-induced hydrodynamic loads. Recognizing that physical constraints may limit a plant's ability to meet these criteria, the staff noted that licensees may take appropriate measures in combination with this option to reduce the potential debris sources in the drywell and suppression pool, thereby reducing the required capacity and overall size of the strainers.

2.2 Design Change F1-97-013

The original suppression pool strainers for the Residual Heat Removal (RHR) and Core Spray (CS) system pumps were replaced under modification F1-97-031 during RO 13. RHR Pump Suction Strainers 10F-4A & -4B and Core Spray Pump Suction Strainers 14F-2A & -2B were replaced with larger, high capacity stainless steel strainers with substantially higher debris loading capacity. The original wire mesh strainers for RHR and CS, installed over the mitered ends of the torus penetration nozzles for each pump train, were replaced with a stacked disk design connected to new flanges on the penetration nozzles. The new, larger strainers extend circumferentially into the torus bays. The surface area of the original and replacement strainers for RHR and CS are given in Table 2.1 for comparison. The new 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 ORIGINAL STRAINERS (ft^2)	SURFACE AREA REPLACEMENT 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. 7.6) specified the design requirements for the replacement strainer assemblies. Additionally, various analyses/calculations were 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 removed the original suction strainers from the suppression pool, modified the torus penetration nozzles to accept new flanges, installed new strainer supports on adjacent torus ring girders and modified supports on the CS suction piping outside containment. No reinforcement of the shell penetrations was required.

Strainer installation was implemented with the torus suppression pool drained. Technical Specification 3.5.F provided ECCS operability requirements for the installation. The strainers were lowered into the torus through the Torus Access Hatch Penetration 16X-200A (RS#5). A pre-approved rigging plan was used to control movement of the strainer assemblies from the Reactor Building Track Bay to the final installed positions in the torus (Ref. 7.7). Load paths and rigging details were defined on Engineering approved sketches. The plan and associated calculations evaluated rigging and safe load paths consistent with criteria in NUREG-0612 (Ref. 7.8).

Post modification examinations were 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 was performed to verify that the suction flow path to each pump is not obstructed and that pump performance satisfies Technical Specification surveillance (Ref. 7.10) and IST acceptance criteria for flow and differential pressure.

A temporary construction opening was cut in the torus to facilitate personnel and equipment movement during strainer installation. The opening was made below the water line in the "A" Bay of the torus once the suppression pool was drained and was repaired following completion of strainer installation. The integrity of the repaired opening was verified by performance of a Primary Containment Pressurization Test (Ref. 7.49). The test pressurized 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 provided a structural load on the torus chamber to verify integrity had 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. 7.53). The NRC Safety Evaluation for the submittal and authorization for the alternative test method were documented in Ref. 7.54.

2.3 Design and Licensing Bases

Original design and licensing documentation do not include any requirement to quantify debris generation in the drywell following a design basis LOCA or to ensure adequate ECCS pump NPSH can be maintained with postulated debris transported to the suppression pool. The original design basis for RHR and Core Spray suction strainers, as defined in the General Electric Design Specifications (Ref. 7.18, 7.19), specified a strainer design that would ensure minimum NPSH requirements could be maintained with the strainers 50% plugged. A review of the original FSAR and AEC Safety Evaluation Report for JAF (Ref. 7.11) and current plant licensing basis documentation found no commitment regarding suction strainer blockage other than to the overall objective of providing adequate NPSH for all ECCS pumps under all operating conditions including the most limiting design basis accident (Ref. 7.9e).

Although installation of the high capacity RHR and Core Spray suppression pool suction strainers was completed during RO 13, existing plant design and licensing bases were not updated to reflect debris generation and transport criteria. In September 1998, immediately preceding RO 13, NYPA extended its commitment to confirm completion of actions taken relative to the Bulletin for, at most, one refueling outage to allow a small quantity of microporous insulation to remain in the drywell pending additional analysis and testing (Ref. 7.20). This action was justified by the significant dose savings which could be realized by not performing the work required to replace the microporous insulation.

Debris quantities used in the performance analysis supporting strainer design did not include a microporous insulation component although this type of insulation is installed on pipe whip restraints in the drywell. Since experimental evidence available during the design phase suggested that microporous insulation debris without fibrous debris could cause significant strainer head loss, and no experimental data was available for microporous insulation debris in combination with large quantities of fibrous debris (as would be expected during a postulated

LOCA at JAF), project analyses assumed all microporous insulation would be removed during Refueling Outage 13.

Following start-up from RO 13, a series of tests was conducted at the Alden Research Laboratory (ARL) to characterize the impact of microporous insulation on the head loss performance of the newly installed ECCS suction strainers. The tests demonstrated that the head loss for a mixed fibrous insulation/microporous insulation debris bed could be quantified by the same correlations used for other particulate debris components if the microporous to fiber mass ratio does not exceed 20%. Revision of debris generation and strainer head loss analyses incorporating a new source term which includes some microporous insulation have confirmed that NPSH margin can be maintained under design basis conditions. The analyses were used to identify specific pipe whip restraints for which replacement of microporous insulation would be required. This work is scheduled for RO 14 under a separate design change package (Ref. 7.58) and is tracked as Action Item 5.4.

Replacement of the microporous insulation during RO 14 will complete actions required to address issues cited in Bulletin 96-03. The design basis for ECCS suction strainers will be updated to reflect criteria documented in design change package F1-97-013. These changes will be documented in the FSAR reflecting completion of the licensing commitments made for resolution of the debris generation and transport concerns raised in the Bulletin.

2.4 Revised Containment Response Analysis

The primary containment long term response to a Loss-Of-Coolant Accident is described in FSAR Section 14.6 (Ref. 7.9j) for the initial core. Power Uprate and two successive increases in ultimate heat sink temperature significantly increased peak suppression pool temperature since the initial core. Following installation of the replacement ECCS pump suction strainers during RO 13, General Electric re-assessed the long-term containment response for a recirculation line break using plant unique values for decay heat, RHR heat exchanger fouling factors, ECCS flow rates, and other relevant data, for evaluation of ECCS NPSH margin.

The revised containment analysis in GE-NE-T23-00766-00-01 (Ref. 7.56) evaluated containment response with, among other plant-specific factors, current rated reactor power (2587 MWt) and licensing basis ultimate heat sink temperature (85°F), both of which drive peak suppression pool temperature and airspace pressure higher, and a larger RHR heat exchanger K-value which significantly increased heat exchanger effectiveness, decreasing peak suppression pool temperature and airspace pressure. The net result was a lower peak suppression pool temperature (196.3°F) and corresponding suppression chamber airspace pressure (22.5 psia) for ECCS pump NPSH evaluation than that assumed in the original plant licensing basis for the design basis LOCA (FSAR Tables 14.6-1 and 14.6-2). The lower peak suppression pool temperature also impacts the net positive suction head required (NPSHR) curves documented in FSAR Figures 6.5-1A & B. These changes will be incorporated into the plant licensing basis by this NSE.

3.0 SAR REVIEW

3.1 A FOLIO search of the JAF FSAR (Ref. 7.9) and Technical Specifications (Ref. 7.10) 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, 3.2.4, 4.8, 5.2, 6.1, 6.2, 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. 7.13)
- NEDC-32016P-1, Rev.1, Power Uprate Safety Analysis for the James A FitzPatrick Nuclear Power Plant, prepared by the General Electric Company (Ref. 7.14)
- NSE JAF-SE-96-048, Rev. 2, Revision to FSAR to Raise Maximum Allowable Lake Temperature form 82°F to 85°F (Ref. 7.15)
- NSE JAF-SE-98-006, Rev. 0, FitzPatrick Cycle 14 Reload Core (Ref. 7.55)
- 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. 8

3.2 The following sections of the FSAR were revised following RO 13 to incorporate the 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 The following sections of the FSAR will require revision following RO 14 to incorporate changes to the plant design and licensing bases for the ECCS suction strainers:

- Section 6.2.9 lists the physical effects of the design basis LOCA which have been addressed in the ECCS design bases. Revision of this section is required to include debris generation and transport to the suppression pool, and hydrodynamic loads in the suppression pool.
- Section 6.4.3, The description of criteria used for qualification of Core Spray equipment, piping and supports should be amended to state that Core Spray piping and supports located in the suppression pool have been designed and analyzed for postulated DBA LOCA hydrodynamic loads.
- Section 6.4.4, The description of criteria used for qualification of LPCI pumps and piping should be amended to state that RHR equipment, piping and supports located in the suppression pool have been designed and analyzed for postulated DBA LOCA hydrodynamic loads.
- Section 6.5.1 currently describes the factors considered in calculating NPSH for the ECCS pumps. This discussion should be revised to include head loss across suction strainers due LOCA-generated debris blockage.
- Section 6.5.1 also discusses compliance with Safety Guide No. 1 and the need to credit containment overpressure for RHR and Core Spray NPSH. The October 1999 containment

analysis in GE-NE-T23-00766-00-01 (Ref. 7.56) significantly decreased the peak long-term suppression pool temperature. Re-evaluation of RHR and Core Spray pump NPSH, with the decrease in analytical peak suppression pool temperature to 196.3°F, has demonstrated adequate NPSH without credit for containment overpressure. The discussion in Sectio 6.5.1 should be revised to reflect updated long-term containment conditions and ECCS pump NPSH requirements.

- Figures 6.5-1A & B show long-term transient suppression chamber airspace pressure curves (containment overpressure available) for a design basis LOCA and NPSH required (NPSHR) curves for RHR and Core Spray pumps during the event. As discussed in Section 4.5 below, the NPSHR curves currently documented in Figures 6.5-1A & B reflect the approved licensing basis for the credit of long-term containment overpressure in evaluating RHR and Core Spray NPSHR. The RHR and Core Spray NPSHR curves in Figures 6.5-1A & B should be revised to reflect the updated NPSHR curves. A third curve should be added to each figure to document the licensing basis for containment overpressure in addition to NPSHR and containment overpressure available.

3.4 The following changes to Technical Specification (Bases) are required:

- Technical Specification Bases 3.7 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". Re-evaluation of RHR and Core Spray pump NPSH, with the decrease in analytical peak suppression pool temperature to 196.3°F, demonstrated adequate NPSH without credit for containment overpressure. The Bases should be updated to reflect the current analysis.

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 Debris Generation and Transport

The maximum quantity of debris that could be generated and transported to the suppression pool as a result of a LOCA was calculated in DE&S calculation A384.F02-02 (Ref. 7.16) using criteria developed in the BWR Owners' Group Utility Resolution Guidance document (URG) (Ref. 7.5). The analysis considered debris sources from fibrous piping insulation, metallic foil (RMI) from insulation installed on the reactor vessel, suppression pool sludge, particulate debris (e.g. paint chips, dirt, rust) and other miscellaneous debris. The evaluation examined potential break locations in high energy piping within the drywell, the shape and size of the Zone of Influence (ZOI) for each type of insulation material and the quantity of insulation located within the ZOI.

The analysis in DE&S calculation A384.F02-02 evaluated breaks at all weld locations on high energy piping 12" and larger to ensure the most severe postulated LOCA was considered in accordance with the criteria of Reg. Guide 1.82, Revision 2. A relevant destruction and transport factor was then applied to estimate the amount of debris that would be transported to the suppression pool.

The analysis was revised following design and installation of the strainers during RO 13 to incorporate the new source term for remaining microporous insulation.

4.2 Revised Containment Response Analysis

During the long term containment response (after depressurization of the reactor vessel is complete), the suppression pool is assumed to be the only heat sink in the primary containment. The primary containment long term response to a Loss-Of-Coolant Accident was summarized for the initial core in FSAR Tables 14.6-1 and 14.6-2 for four cases, representing various containment spray and cooling combinations. Containment response was analyzed for the following containment cooling modes, assumed to be initiated by the control room operator 600 seconds after the accident:

Case A

Operation of both RHR system cooling loops delivering 23,100 gpm in the containment cooling mode with four RHR pumps, four RHR service water pumps and 2 RHR heat exchangers, and one Core Spray pump injecting 4,625 gpm. The RHR pumps draw suction from the suppression pool and pump water through the RHR system heat exchangers into the containment via the containment sprays.

Case B

Operation of one RHR system cooling loop with two RHR pumps delivering 11,500 gpm in the containment cooling mode, with two RHR service water pumps and one RHR system heat exchanger, and one Core Spray pump injecting 4,625 gpm. As in Case A, the RHR pumps draw suction from the suppression pool and pump water through the RHR system heat exchangers into the containment via the containment sprays.

Case C

Operation of one RHR system cooling loop with one RHR pump delivering 7700 gpm in the containment cooling mode, two RHR service water pumps and one RHR system heat exchanger, and one Core Spray pump injecting 4,625 gpm. As in the previous cases, the RHR pump draws suction from the suppression pool and pumps water through the RHR system heat exchanger into the containment via the containment sprays.

Case D

Operation of one RHR system cooling loop with one RHR pump delivering 7700 gpm in the LPCI mode, two RHR service water pumps and one RHR system heat exchanger, and one Core Spray pump injecting 4,625 gpm. In this case, the RHR pump draws suction from the suppression pool and discharges flow through the RHR heat exchanger where it is cooled and then injected into the reactor vessel as LPCI flow.

The analysis summarized in Tables 14.6-1 and 14.6-2 showed peak suppression pool temperature for cases C and D as 204°F with corresponding airspace pressures of 12.0 and 15.5 psig, respectively. Core Power Uprate to 2587 MWt and increases in ultimate heat sink temperature to 85°F significantly increased peak suppression pool temperature following a LOCA. The replacement ECCS suppression pool suction strainers were designed for a peak suppression pool temperature of 211.4°F which reflected these changes in plant operating parameters.

Following installation of the replacement strainers during RO 13, General Electric re-assessed the long-term containment response for a recirculation line break using plant unique values for decay heat, RHR heat exchanger fouling factors, ECCS flow rates, and other relevant data. The revised containment analysis in GE-NE-T23-00766-00-01 (Ref. 7.56) significantly reduced the analytical peak suppression pool temperature and suppression chamber pressure for evaluation of ECCS pump NPSH margin. The GE analysis evaluated a case analogous to Case C in FSAR 14.6 for the limiting single failure, i.e. a 100% recirculation suction line break with a failure of one of the redundant emergency power sources.

The analysis assumed the following operating alignments after the initial 600 seconds of a design basis LOCA:

Operation of one RHR system cooling loop with one RHR pump delivering 7500 gpm in the containment cooling mode, two RHR service water pumps and one RHR system heat exchanger, and one Core Spray pump injecting 4100 gpm. The RHR pump draws suction from the suppression pool and pumps water through the RHR system heat exchanger into the containment via the containment sprays.

The revised containment analysis calculated peak long-term suppression pool temperature as 196.3°F with a corresponding suppression chamber airspace pressure of 22.4 psia (7.8 psig) for use in determining ECCS pump NPSH margin. Since NPSH is directly affected by vapor pressure of the liquid being pumped and vapor pressure is strongly influenced by temperature, the reduction in calculated suppression pool temperature from the 211.4°F assumed for ECCS suction strainer design to 196.3°F adds approximately 9.4 feet of head to the NPSH margin. Similarly, the reduction from the peak suppression pool temperature of 204°F (Tables 14.6-1 and 14.6-2) to 196.3°F adds approximately 4.5 feet of NPSH margin to that calculated for the initial core.

4.3 Net Positive Suction Head (NPSH) Evaluation used in Strainer Design

General Electric Design Specifications 22A1472 (Ref. 7.18) and 22A1435 (Ref. 7.19) 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. 7.21) assuming a 50% reduction in strainer surface area to account for debris loading consistent with the original design basis, and for design debris loading in accordance with the criteria of Reg. Guide 1.82, Revision 2. Worst case debris loading for each set of strainers was used to determine a debris head loss value which, when added to the clean strainer term, yielded the fully loaded strainer pressure drop. Debris head loss values were determined in DE&S calculation A384.F02-06, Rev. 1 (Ref. 7.17) based on correlations presented in NUREG/CR-6224 and analysis of testing performed for the PCI stacked disk strainers at the EPRI strainer test facility.

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 NPSH analysis used the maximum long-term suppression pool temperature for the current plant licensing basis, calculated from data documented previously in Safety Evaluations for power uprate and increased ultimate heat sink, NEDC-32016P-1 (Ref. 7.14) and JAF-SE-96-048 (Ref. 7.15), 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 long-term case flow rates used in the NPSH analysis supporting strainer design reflected 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.

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. 7.15, 7.22) 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. 7.24). 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. 7.14, 7.23). Conditions evaluated were specified in Procurement Specification JAF-SPEC-MISC-02871 (Ref. 7.6).

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

4.4 NPSH Re-evaluation (Post Installation)

RHR and Core Spray pump suppression pool NPSH were re-evaluated in Proto-Power calculation 98-019 Revision D for the revised debris mix (including a mircoporous insulation component) and revised long-term containment analysis conditions (Ref. 7.56). Two cases were evaluated in 98-019 Revision D corresponding to the system parameters provided in JAF-ICD-MULTI-03554 (Ref. 7.57) for ECCS pump operation:

Case A

This case is analogous to Case C summarized in FSAR Tables 14.6-1 and 14.6-2 and represents the new design basis for RHR and Core Spray ECCS operation. This case assumes the limiting single failure (failure of an emergency bus), flow rates for LPCI, Containment Spray and Core Spray injection selected to be consistent with other design basis LOCA analyses, and the peak long-term suppression pool temperature calculated in GE-NE-T23-00766-00-01 (Ref. 7.56). Strainer debris head loss values were determined in DE&S calculation A384.F02-06, Revision 2, to reflect the impact of some microporous insulation in the debris mix. Pump flow and suppression pool temperature values used in the NPSH analysis are summarized in Table 4-2 below.

TABLE 4.2
CASE A
NEW DESIGN BASIS NPSH EVALUATION INPUT VALUES - (Limiting Single Failure)

TIME (sec)	SYSTEM	PUMPS PER TRAIN	NUMBER OF TRAINS	FLOW PER TRAIN (gpm)	PEAK SUPPRESSION POOL TEMP (°F)
≤600	RHR (LPCI)	1	2	10500	150
	CS	1	1	6800	
40,764	RHR (Containment Spray)	1	1	7600	196.3
	CS	1	1	4200	

Case B

This case is analogous to Case A summarized in FSAR Tables 14.6-1 and 14.6-2 and represents ECCS operation with no single failure assumed. Unlike Case A in the FSAR tables, this case assumes both Core Spray pumps operate throughout the duration of the event. This case was evaluated in JAF-ICD-MULTI-03554 (Ref. 7.57) to measure suppression pool temperature sensitivity to additional pump heat. As for the design basis case, strainer debris head loss values from DE&S calculation A384.F02-06, Revision 2, were used in the NPSH analysis. Pump flow and suppression pool temperature values used in the NPSH analysis are summarized in Table 4-3 below.

TABLE 4.3
CASE B
NPSH EVALUATION INPUT VALUES - (No Single Failure)

TIME (sec)	SYSTEM	PUMPS PER TRAIN	NUMBER OF TRAINS	FLOW PER TRAIN (gpm)	PEAK SUPPRESSION POOL TEMP (°F)
≤600	RHR (LPCI)	2	2	20800	150
	CS	1	2	6800	
Long Term	RHR (Core Spray)	2	2	15700	163
	CS	1	2	4200	

4.5 Containment Overpressure

The NPSH analysis supporting strainer design (Ref. 7.21) included 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. 7.13) 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. 7.25). 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 was discussed in NYPA's response to Generic Letter 97-04 (Ref. 7.26). The response letter included containment pressure curves updated to reflect the influence of these two design changes on containment response. As on the original curve for RHR, the revised curves show the period during which overpressure is required for RHR and Core Spray NPSH to be coincident with the time at which maximum pressure is postulated in the wetwell The reduction in NPSH due to temperature is offset by increased containment pressure. FSAR Figures 6.5-1A&B were updated per NSE JAF-SE-96-048 to reflect the current plant operating conditions.

In a December 4, 1998 letter regarding JAF's response to Bulletin 96-03 (Ref. 7.27), the NRC documented its review of the basis for use of containment overpressure in calculation of available RHR and Core Spray pump NPSH at JAF. The letter confirmed the NRC's approval for use of containment overpressure in the JAF design basis. The plant, it noted, "was licensed to credit less than 2 psig of containment overpressure for the RHR for the time period presented in Figure Q.6.4-1 of Supplement 4 of the FitzPatrick FSAR. Additionally, staff approval of power uprate and Amendment 239 of the FitzPatrick Technical Specifications allowed credit of less than 2 psig of containment overpressure for the core spray pumps."

The letter further clarifies the NRC's position in stating that "The staff notes that it would consider any future increase in reliance of containment overpressure credit, above 2 psig, for the RHR and/or core spray pumps to be an unreviewed safety question, which would require NRC approval to amend the operating license before the licensee could make such a change. This would also include any increase or change in time the containment overpressure is required, i.e., a longer period than currently depicted in the GL 97-04 response figures and as described in FSAR Section 6.5.1, or short-term requirements."

The analyses developed in support of the design change demonstrated that replacement of the RHR and Core Spray suppression pool suction strainers could be implemented without impact on the time and duration for which overpressure would be required. Proto-Power calculation 98-019, Revision C (Ref. 7.21) concluded that NPSH margin would be maintained with 50% of the strainer surface area blocked or with design debris loading assuming credit for no more than 2 psi containment overpressure. The re-evaluation of RHR and Core Spray pump NPSH in Proto-Power calculation 98-019, Revision D, which incorporated revised containment analysis values from GE-NE-T23-00766-00-01 (Ref. 7.56), showed that, with the decrease in analytical peak suppression pool temperature to 196.3°F, adequate NPSH can be demonstrated for the RHR and Core Spray pumps without credit for containment overpressure.

The long-term containment pressure and NPSHR curves for the RHR and Core Spray pumps in FSAR Figures 6.5-1A&B will be updated to reflect the peak long-term suppression pool temperature determined in GE-NE-T23-00766-00-01. Additionally, the revised NPSHR curves will incorporate head loss through the newly installed high capacity suppression pool suction strainers with limiting debris deposition (including a microporous component). The NPSHR curves currently shown in the figures will be re-labeled as representing the approved plant licensing basis for credit of containment overpressure in NPSH evaluation.

4.6 Strainer Hole Sizing

General Electric Design Specifications 22A1472 (Ref. 7.18) and 22A1435 (Ref. 7.19) 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. 7.28) 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 used 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.

The Cycle 14 core reload safety evaluation (Ref. 7.55) and FSAR Section 3.2.4 (Ref. 7.9a) discussed the Debris Filter Lower Tie Plate (DFLTP) flow hole sizes incorporated on Reload 13 fuel bundles. The flow hole size was reduced from 0.26" on Reload 12 fuel to 0.112" in Reload 13. The finer inlet flow holes are intended to reduce the fraction of the types of debris known to have caused debris fretting failures from entering the rodded portion of the fuel bundles. The DFLTP hole sizes are larger than the $^{3}/_{32}$" (0.094") diameter holes in the new ECCS suppression pool suction strainers. This ensures that debris small enough to pass through the torus strainers would not be capable of blocking fuel assembly flow.

4.7 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. 7.29). The possibility for pump degradation or condensation induced water hammer were evaluated in accordance with NUREG-0897 (Ref. 7.30).

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. 7.29) 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 was included in F1-97-031 to add requirements regarding LCO entry during planned operation of a Core Spray pump. This requirement was documented in a revision to FSAR Section 6.6 (Ref. 7.9f) 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. 7.29) 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.

4.8 Vortex Limits

The minimum required water level above the strainers to prevent vortexing was calculated in DE&S calculation A384.F02-03 (Ref. 7.31) 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 demonstrated 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 concluded 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. 7.32) and EOP-3 (Ref. 7.33) Vortex Limits curves for the range of RHR and Core Spray pump flow rates. Existing EOP calculations and the EOP Vortex Limit curves were revised to incorporate minimum submergence data for the new strainers. Revision of the EOP calculation and curves was tracked in F1-97-031.

4.9 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. 7.22). 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. 7.34). 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 did 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.

4.10 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 involved 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 differed from the original plant unique analysis were summarized in a supplement to the FitzPatrick Plant Unique Analysis Report (PUAR) (Ref. 7.45).

The replacement strainers and strainer supports were 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. 7.35) and the plant unique methodologies as specified in the PUAR (Ref. 7.36, 7.37). 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.

4.11 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. 7.38), NUREG-0661 (Ref. 7.35) and the PUAR (Ref. 7.36, 7.37) 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 did 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. 7.39, 7.40). 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. 7.41). 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 s 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 were documented in the PUAR revision (Ref. 7.45)

4.12 <u>Load Combinations</u>

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. 7.42) 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. 7.43). This method has been widely used throughout the industry on Mark I plants. A supplement to the PUAR was prepared, which documents the use of SRSS for FitzPatrick (Ref. 7.45).

4.13 <u>Integral Welded Attachments</u>

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. 7.44), 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 was documented in a supplement to the PUAR (Ref. 7.45).

4.14 Ring Girder Supports

Each RHR strainer assembly is supported by three ring girder supports. A single ring girder support is 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, reinforcement of the ring girders used to support the RHR strainer assemblies was required. 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.

4.15 RHR Bellows Assemblies

The new design for the RHR suction strainers 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 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.

4.16 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. 7.50, 7.51). The valves were evaluated to ensure structural integrity would be maintained under all design conditions (Ref. 7.46). The predicted accelerations acting on the motor operators are bounded by the Limitorque qualification (Ref. 7.52)

4.17 Suppression Pool Inventory

The replacement strainers and ring girder supports displace approximately 300 ft^3 more water volume than the original 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 remain unchanged and, therefore, the modification has 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. 7.47). 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.

4.18 Temporary Torus Construction Opening

A temporary construction opening was cut in the torus to facilitate personnel and equipment movement during strainer installation. The opening was 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 was repaired using the same or equivalent material and weld processes qualified to meet notch toughness requirements of the original construction code. Weld integrity was 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 were performed at the calculated peak drywell pressure.

A qualitative evaluation concluded that the temporary opening would 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 ensured the strength of the welded joint to be equal to or greater than that of the base material. Thus the repair did not affect primary containment structural or pressure boundary integrity.

4.19 Primary Containment Pressurization Test

The integrity of the repaired torus opening was verified by performance of a Primary Containment Pressurization Test after the suppression pool was refilled (Ref. 7.49). The test pressurized the primary containment volume to an internal pressure ≥ 45 psig for a minimum of one hour. The reinstated torus opening, therefore, was subjected to a hydrostatic pressure >45 psig. Visual inspection of the weld for leak-tightness (VT-2) was performed under the hydrostatic conditions in lieu of the pneumatic test required by ASME Section XI, IWE-5221. Additionally, the test provided a load on the torus chamber to allow visual verification of structural integrity at the repaired area. The test pressure was based on the 45 psig peak primary containment internal pressure postulated for the design basis loss of coolant accident (Ref. 7.10).

The procedure developed to pressurize primary containment was 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 were incorporated into the Primary Containment Pressurization Test. Since this post modification test was limited to verification of local leak-tightness and structural integrity in the area of the repair weld, it did not include requirements for pressure maintenance/stabilization beyond the nominal one hour period and did not measure pneumatic leakage through containment penetrations. Plant configuration and conditions established for the post modification test were consistent with those in the periodic leakage rate surveillance. The one time test had no impact on the leakage rate surveillance program described in the FSAR and did not alter the description of any test or procedure in the safety analysis report.

Testing was 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 was not required during the testing period.

4.20 Dose Rates

It is expected that the strainers will accumulate radioactive particulates during normal plant operation in addition to the deposition of debris during an accident. Since the strainers will not generate any additional particulates, offsite dose rates will not be affected. Local dose rates within the torus and external to the torus in the area of the strainers may be impacted during normal plant activities. These would be evaluated and controlled as necessary in accordance with plant procedures to minimize personnel exposure. In an accident scenario, water from the suppression pool is circulated through the systems increasing dose rates in the areas of the pumps. For accident response, it is expected that the dose rate in any area in which operator actions are required would not be affected significantly.

4.21 10CFR 50.59 Evaluation

Replacement of the suppression pool suction strainers for the RHR and Core Spray pumps, cutting a temporary construction opening in the torus, and changes to plant design and licensing bases to incorporate debris generation and deposition on the RHR and Core Spray suction strainers:

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 did not alter any initial condition or assumption used in the accident analysis. FSAR Section 12.2A.6 (Ref. 7.9g) 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 had 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 (Ref. 7.9h) 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 involved 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 (Ref. 7.9i). This modification did not alter the overall configuration of any ECCS pump suction line and added no components that could affect shutdown cooling. The function of the replacement pump suction strainers in the suppression pool is identical to the original components. Installation of strainers with significantly larger surface area and higher debris capacity 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, did 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 objective of the Emergency Core Cooling Systems (ECCS) is to limit, in conjunction with the primary and secondary containments, the release of radioactive materials to the environs following a loss-of-coolant accident (LOCA) so that resulting radiation exposures are kept within the guideline values given in 10 CFR 100. This objective is primarily achieved by maintaining core coolant inventory to prevent fuel damage. (Ref. 7.9a)

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 does 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 or with limiting debris deposition on the strainers, 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 originally installed, increasing the likelihood of long-term pump availability following a LOCA.

In the event of an accident, the 3/32" hole sizes in the replacement strainers will provide improved protection against blockage of lower tie plate flow holes and introduction of debris into the fuel bundles. The possibility of fuel cladding damage, therefore, is reduced by the modification.

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 displace more water volume than the original 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, are unaffected.

The strainers, strainer supports, torus structure, and torus attached piping were 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 were 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 original 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. 7.21) verifies that, with 50% of their suction strainer surface blocked or with the analytical limiting debris mix deposited on the strainers, 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.

Since the modification does not affect thermal input to or heat removal capacity from the suppression pool, $NPSH_{AVAILABLE}$ is not reduced by the design change from that in the existing licensing basis. Additional NPSH margin has been demonstrated analytically by the containment reanalysis in GE-NE-T23-00766-00-01 (Ref. 7.56). With the decrease in peak suppression pool temperature to 196.3°F, adequate NPSH exists for the RHR and Core Spray pumps without credit for containment overpressure.

Air and steam bubble formation in the suppression pool as a result of SRV or LOCA downcomer discharge has been evaluated (Ref. 7.29). 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 were 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. 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 original 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 did not adversely affect primary containment. The removal, repair and post modification testing was 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 was not required during the construction and testing period.

Qualification and NDE of the repair weld ensured the strength of the welded joint will be equal to or greater than that of the base material. Post modification testing (Ref. 7.49) verifed leak-tightness and structural integrity of the repaired area. Testing for the repaired torus opening was performed at the peak primary containment internal pressure for the design basis loss of coolant accident. The 45 psig nominal test pressure was below the 56 psig design of the vessel (Ref. 7.9). The test, therefore, provided 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, did 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. 7.23, 7.56), 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 does 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. 7.29). 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 were 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 original 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 were the original 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 does not adversely affect the integrity of the containment pressure vessel. The strainers were 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 does 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 is not affected nor is the impact of any single failure.

Technical Specification 3.7.1 limitations on suppression pool level 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 is maintained.

Analysis has shown that the 105,900 ft^3 suppression pool water volume currently specified in Technical Specification Bases 3.7 would be met even with the installation of larger RHR and Core Spray suction strainers (Ref. 7.47). The suppression pool water volume required to act as the heat sink for the RCS energy release following a postulated LOCA has not been 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. 7.49) was 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, was not required while the test is in progress.

The test procedure disabled 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 was performed with the plant in Cold Condition, the drywell high pressure scarm function was 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 included steps to ensure suppression pool level remained within Technical Specification 3.7 limits and an RHR loop was maintained in standby. Since the requirement for drywell high pressure initiation could not be met, the test procedure required 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 did not affect the reactor water level signal and did not affect their availability.

Performance of the primary containment pressurization test (Ref.7.49), therefore, did 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 were procured and installed in accordance with the applicable requirements for Category I components. There has been no impact on the Quality Assurance Program.

The changes did not impact the site fire protection program. No fire barrier was affected, no flammable or combustible material was added or removed from any area of the plant, and no fire detection/suppression equipment or emergency lighting was impacted. FPES-04A Exhibit 1, (Ref. 7.48) was prepared, reviewed and approved. The modification was evaluated for impact on the Fire Protection Program and Appendix R Safe Shutdown Analysis and the commitments made therein were not impacted, invalidated, or affected.

Touch-up painting, required at the completion of strainer and support installation, was controlled by existing plant procedures which specify use of pre-approved paints. The modification had no other effect on air or water quality, did not involve use of any hazardous substance, did not affect any property outside the plant buildings, did 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 does not affect the operation or performance of any component or system required for accident mitigation. EOP revisions were 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 have been 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 Completed (See JAF-SE-98-013, Rev. 1)

5.4 Removal of microporous insulation from pipe whip restraints in the drywell at locations delineated in the debris generation and transport analysis, DE&S calculation A384.F02-02, must be completed to validate the assumptions of the Proto-Power Corp. Calculation No. 98-019, Rev. D. The replacement of microporous insulation for the specified restraints is included in the scope of Minor Modification M1-97-131 and is scheduled for implementation during RO 14.

5.5 An Update of Technical Specification Bases 3.7 is required as described in Section 3.4 above. The Bases changes can be implemented upon approval of this NSE revision in accordance with MCM-4, Step 6.1.9.

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", installed new suppression pool suction strainers for Residual Heat Removal and Core Spray system pumps. The replacement strainers are substantially larger than the original 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, but with the worst case debris loading determined in accordance with the criteria of Reg. Guide 1.82, Revision 2. 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 re-evaluation of the long-term suppression pool temperature following a design basis Loss-of-Coolant-Accident, developed by General Electric for FitzPatrick, demonstrated additional ECCS pump NPSH margin. The new containment analysis in GE-NE-T23-00766-00-01 decreased the analytical peak suppression pool temperature to 196.3°F.

A temporary construction opening was made in the torus to facilitate personnel and equipment movement for the strainer installation. Integrity of the repaired opening was ensured by qualification and non-destructive examination of the weld, and by post modification testing. A primary containment pressurization test was 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 has not been 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 USNRC Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors, May 6, 1996

7.3 NUREG/CR-6224, Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris, October 1995

7.4 Regulatory Guide 1.82, Rev. 2, Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident, May 1996

7.5 NEDO-32686, Draft, Utility Resolution Guidance for ECCS Suction Strainer Blockage, prepared by the Boiling Water Reactor Owners' Group, November 1996

7.6 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

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

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

7.9 FSAR, 1999 FSAR Update

a. Section 3.2.4, Fuel Description
b. Section 6.1, Emergency Core Cooling System Safety Objective
c. Section 6.2, Emergency Core Cooling System Safety Design Bases
d. Section 6.4, Emergency Core Cooling System Safety Description
e. Section 6.5, Emergency Core Cooling System Safety Evaluation
f. Section 6.6, Emergency Core Cooling System Inspection and Testing
g. Section 12.2A.6, Major Accidents
h. Section 12.2A.7, Safe Shutdown Evaluation
i. Section 14.5, Analysis of Abnormal Operational Transients and Reactor Vessel Overpressure
j. Section 14.6, Analysis of Design Basis Accidents

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

7.11 U.S. Atomic Energy Commission Safety Evaluation of the James A FitzPatrick Nuclear Power Plant, Docket No. 50-333, November 20, 1972

7.12 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Bulletin 96-03 Boiling Water Owners Group Topical Report NEDO-32686 "Utility Resolution Guidance for ECCS Suction Strainer Blockage" Docket No. PROJ0691, 12/30/97 Draft

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

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

7.15 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

7.16 Duke Engineering & Services Calculation No. A384.F02-02, Rev. 4, JAF Nuclear Plant: Estimation of Debris Generation and Transport to the Suppression Pool following a LOCA

7.17 Duke Engineering & Services Calculation No. A384.F02-06, Rev. 1, Strainer Head Loss Performance Analysis (ITS)

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

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

7.20 Michael J. Colomb, JAF Site Executive Officer, letter to USNRC, JAFP-98-0306, Information Regarding NRC Bulletin 96-03, September 16, 1998

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

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

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

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

7.25 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

7.26 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

7.27 Joseph F Williams, U.S. Nuclear Regulatory Commission, letter to J. Knubel, New York Power Authority, James A. FitzPatrick Nuclear Power Plant - NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors" (TAC No. M96-146), December 4, 1998

7.28 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

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

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

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

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

7.33 Emergency Operating Procedure EOP-3, Rev. 4, Failure to SCRAM

7.34 NUREG-0783, Suppression Pool Temperature Limits for BWR Containments, November 1981

7.35 NUREG-0661, Safety Evaluation Report Mark I Containment Long-Term Program, Resolution of Generic Technical Activity A-7, July 1980, including Supplement 1, August 1982

7.36 Teledyne Engineering Services, Technical Report TR-5321-1, Mark I Containment Program Plant Unique Analysis Report of the Torus Suppression Chamber for James A. Fitzpatrick Nuclear Power Plant, Rev. 1, November 1984

7.37 Teledyne Engineering Services, Technical Report TR-5321-2, Mark I Containment Program Plant Unique Analysis Report of the Torus Attached Piping for James A. Fitzpatrick Nuclear Power Plant, Rev. 1, November 1984

7.38 NEDO-21888, Rev. 2, November 1988, General Electric, Mark I Containment Program Load Definition Report

7.39 DE&S Test Report No. TR-ECCS-GEN-01-NP, Rev. 2, Test Report for Hydrodynamic Inertial Mass Testing of ECCS Suction Strainers

7.40 DE&S Test Report No. TR-ECCS-GEN-05-NP, Rev. 1, Supplement 1 to Hydrodynamic Inertial Mass Testing of ECCS Suction Strainers - Free Vibration Analysis

7.41 DE&S Test Report No. TR-ECCS-GEN-011, Rev. 0, ECCS Suction Strainer Hydrodynamic Test Summary Report

7.42 GE Report NEDE-24632, Mark I Containment Program - Cumulative Distribution Functions for Typical Dynamic Responses of a Mark I Torus and Attached Piping Systems, December 1980

7.43 D.B. Vassallo (USNRC) letter to H. C. Pfefferlen (GE), Acceptability of SRSS Method for Combining Dynamic Responses in Mark I Piping Systems, March 10, 1983

7.44 NEDO-24583-1, "Mark I Program, Structural Acceptance Criteria, Plant Unique Analysis Application Guide," October 1979

7.45 JAF-RPT-MULTI-03000, Rev. 0, ECCS Suction Strainer Replacement Modification Supplement to the Plant Unique Analysis Report

7.46 Altran Calculation No. 93179-C-60, Rev. 0, Valve Thrust Assessment 16" Powell Gate Valve: 14MOV-7A & B, June 1998

7.47 JAF-CALC-MISC-02924, Rev. 0, Volumetric Displacement of Water within Torus Due to Submerged Structures

7.48 ESM: FPES-04A, Rev. 1, Fire Protection/Appendix R Compliance Procedure

7.49 Temporary Surveillance Test, TST-87, Rev. 0, Primary Containment Pressurization Test

7.50 Duke Engineering & Services Calculation No. A384.F02-10, Rev. 2, Core Spray Penetration X-227A TAP Piping Reanalysis for the Replacement Suction Strainer Assemblies

7.51 Duke Engineering & Services Calculation No. A384.F02-11, Rev. 2, Core Spray Penetration X-227B TAP Piping Reanalysis for the Replacement Suction Strainer Assemblies

7.52 Limitorque Report B0115, Hydrodynamic Vibration Test (New Loads), 6/24/82

7.53 J. Knubel, New York Power Authority, letter to U.S. Nuclear Regulatory Commission, JPN-98-037, Request for Approval of Relief Request No. RR-16 Alternative Testing per 10 CFR 50.55a(a)(3)(I) for 1992 Edition ASME Section XI, IWE-5221, Testing Following Repair, Modification or Replacement, August 7, 1998

7.54 S. Singh Bajwa, U.S. Nuclear Regulatory Commission, letter to J. Knubel, New York Power Authority, James A. FitzPatrick Nuclear Power Plant - Alternative Testing of Containment Following Emergency Core Cooling System (ECCS) Suction Strainer Replacement (TAC No. MA2472), September 21, 1998

7.55 JAF-SE-98-006, Rev. 0, Cycle 14 Reload Core

7.56 GE-NE-T23-00766-00-01, James A. FitzPatrick Nuclear Power Plant Containment Analysis, October 1999

7.57 NYPA Interface Control Document No. JAF-ICD-MULTI-03554, Revision 0, ECCS Suction Strainer Project – Revised Design Inputs for Strainer Performance and NPSH Evaluation

7.58 JAF Minor Modification M1-97-131, Rev. 0, Drywell Pipe Mineral Wool Insulation Replacement and Drywell Pipe Rupture (Whip) Restraint Insulation Replacement

Attachment 2 to JAFP-00-0288

Nuclear Safety Evaluation JAF-SE-98-025

High Pressure Coolant Injection and Reactor Core Isolation Cooling Suppression Pool Suction Strainer Replacement

Nuclear Engineering
NUCLEAR SAFETY EVALUATION FORM

☐IP3 ☒ JAF Nuclear Station NSE Number: JAF-SE-98-025 Revision #: 2 Full Rev: ☒; Partial Rev. ☐ (See Note 1 below)

Activity Number: F1-98-100 Activity: ☒ Modification ☐ Procedure ☐ Test ☐ Experiment ☐ Other

Title: High Pressure Coolant Injection and Reactor Core Isolation Cooling 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 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. *[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.

Prepared by: A.L. Krinzman *A.L. Krinzman* Date: 7/26/00

Reviewed by: T.M. Driscoll *Thomas M. Driscoll* Date: 7/31/00

Recommended: Approval ☑; AZ Disapproval ☐; PORC Meeting 00-053 Date: 8/15/00

Approved by: *[signature]* Date: 8/15/00
Site Executive Officer or Designee

Distribution: SRC Coordinator, JAF Dir Design Eng/IP3 Systems Engineering Mgr (annual 50.59 report) RMS-JAF/IP3 Preparer

1.0 PURPOSE

The original High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) suppression pool suction strainers were replaced by plant modification F1-98-100 during Refuel Outage (RO) 13 (Ref. 7.1). The replacement strainers have greater debris loading capacity to better accommodate the debris generated as a result of a small or intermediate break Loss of Coolant Accident and smaller openings for improved filtration.

Since the original design basis documentation for HPCI and RCIC did not explicitly define blockage criteria for the suppression pool suction strainers, the current design basis assumed for installation of replacement HPCI and RCIC strainers was the 50% blockage requirement specified for Residual Heat Removal and Core Spray.

Revision 2 addresses changes to the plant design and licensing bases for the HPCI and RCIC suction strainers to incorporate debris blockage criteria outlined in NRC Bulletin 96-03 (Ref. 7.2).

2.0 DESCRIPTION OF PROPOSED ACTIVITY

2.1 NRC Bulletin 96-03

NRC Bulletin 96-03 (Ref. 7.2) notified licensees of the operating events and detailed analysis, documented in NUREG/CR-6224 (Ref. 7.3), which demonstrated the potential for common-cause failure due to excessive buildup of debris from thermal insulation, corrosion products and other particulates on ECCS pump strainers. The NRC staff concluded that this issue must be resolved to ensure compliance with 10 CFR 50.46 for long-term core cooling capability following a LOCA. Noting that Regulatory Guide 1.82, Revision 2 (Ref. 7.4) provided an acceptable method of ensuring compliance with 10 CFR 50.46, all BWR licensees were requested to implement "appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA".

The Bulletin requested implementation of corrective actions by the end of the first refueling outage starting after January 1, 1997. The NRC staff identified three potential resolution options in the Bulletin but allowed alternative solutions which could be demonstrated to provide an equivalent level of assurance that the ECCS will be able to perform its safety function following a LOCA. Consequently, the BWR Owners' Group developed a Utility Resolution Guidance (URG) document (Ref. 7.5) to provide detailed guidance on performance of plant-specific analyses consistent with Reg. Guide 1.82, Revision 2.

JAF selected implementation of a design change, designated as Option 1 in the Bulletin, by which high capacity, passive strainers would be installed in the suppression pool for each ECCS pump. Design considerations cited in the Bulletin for this option include providing a strainer design with sufficient capacity to ensure that debris loadings for the most severe postulated LOCA do not cause a loss of NPSH for the ECCS and a design which ensures structural integrity for the strainers under LOCA-induced hydrodynamic loads. Recognizing that physical constraints may limit a plant's ability to meet these criteria, the staff noted that licensees may take appropriate measures in combination with this option to reduce the potential debris sources in the drywell and suppression pool, thereby reducing the required capacity and overall size of the strainers.

2.2 Design Change F1-98-100

The original suppression pool strainers for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system pumps were replaced under modification F1-98-100 during RO 13. HPCI Pump Suction Strainer 23F-9 and RCIC Pump Suction Strainer 13F-8 were replaced with larger, high capacity stainless steel strainers with substantially higher debris loading capacity. The original wire mesh strainers for HPCI and RCIC were installed over the mitered ends of the torus penetration nozzles for each pump. These were replaced with Performance Contracting Incorporated (PCI) Sure-Flow™ stacked disk strainers which connect to new flanges on the penetration nozzles and extend axially into the torus bays. The new design provides high debris loading capacity, low flow entrance velocities and, therefore, low strainer head losses. The surface area of the original and replacement strainers for HPCI and RCIC are given in Table 2.1 for comparison.

TABLE 2.1

STRAINER SURFACE AREA COMPARISON

PUMP	STRAINER ID	PEN NO.	PEN SIZE (in)	SURFACE AREA ORIGINAL STRAINERS (ft^2)	SURFACE AREA REPLACEMENT STRAINERS (ft^2)
HPCI 23P-1B	23F-9	X-226	16	6.6	133
RCIC 13-P1	13F-8	X-224	6	1.3	10

Procurement Specification JAF-SPEC-MISC-02871 (Ref. 7.6) specified the design requirements for the replacement strainer assemblies. Additionally, various analyses/calculations were prepared to demonstrate the acceptability of the available NPSH, structural integrity of the strainers, piping supports and penetrations, hydraulic flow within the suppression pool, and the heavy load path for installation. (A complete list is included in F1-98-100, Section 10.0) To support installation of the new strainers, the modification removed the original suction strainers from the suppression pool, modified the torus penetration nozzles to accept new flanges, and modified supports on the HPCI suction piping outside containment. No reinforcement of the torus penetrations was required.

Strainer installation was implemented with the torus suppression pool drained. Technical Specification 3.5.F (Ref. 7.7) provided ECCS operability requirements for the installation. The strainers were lowered into the torus through the Torus Access Hatch Penetration 16X-200A (RS#5). A pre-approved rigging plan was used to control movement of the strainer assemblies from the Reactor Building Track Bay to the final installed positions in the torus (Ref. 7.8). Load paths and rigging details were defined on Engineering approved sketches. The plan and associated calculations evaluated rigging and safe load paths consistent with criteria in NUREG-0612 (Ref. 7.9).

Post modification examinations were performed to ensure the structural integrity of the strainer assembly components and torus attached piping affected by this modification, in accordance with applicable portions of the ASME code. Post modification testing verified an unobstructed flow path from the suppression pool to the HPCI and RCIC pumps.

2.3 Design and Licensing Bases

Original design and licensing documentation did not include any requirement to quantify debris generation in the drywell following a design basis LOCA or to ensure adequate ECCS pump NPSH can be maintained with postulated debris transported to the suppression pool.

Since the original design basis documentation for HPCI and RCIC did not explicitly define blockage criteria for the suppression pool suction strainers, the current design basis assumed for installation of replacement strainers was the 50% blockage requirement specified for Residual Heat Removal and Core Spray. The original design basis for RHR and Core Spray suction strainers, as defined in the General Electric Design Specifications (Ref. 7.15, 7.16), specified a strainer design that would ensure minimum NPSH requirements could be maintained with the strainers 50% plugged. A review of the original FSAR and AEC Safety Evaluation Report for JAF (Ref. 7.11) and current plant licensing basis documentation found no commitment regarding suction strainer blockage other than to the overall objective of providing adequate NPSH for all ECCS pumps under all operating conditions including the most limiting design basis accident.

Although installation of the high capacity HPCI and RCIC suppression pool suction strainers was completed during RO 13, the existing plant design and licensing bases were not updated to reflect debris generation and transport criteria. In September 1998, immediately preceding RO 13, NYPA extended its commitment to confirm completion of actions taken relative to the Bulletin for, at most, one refueling outage to allow a small quantity of microporous insulation to remain in the drywell pending additional analysis and testing (Ref. 7.17). This action was justified by the significant dose savings which could be realized by not performing the work required to replace the microporous insulation.

Debris quantities used in the performance analysis supporting strainer design did not include a microporous insulation component although this type of insulation is installed on pipe whip restraints in the drywell. Since experimental evidence available during the design phase suggested that microporous insulation debris without fibrous debris could cause significant strainer head loss, and no experimental data was available for microporous insulation debris in combination with large quantities of fibrous debris (as would be expected during a postulated LOCA at JAF), project analyses assumed all microporous insulation would be removed during Refueling Outage 13.

Following start-up from RO 13, a series of tests was conducted at the Alden Research Laboratory (ARL) to characterize the impact of microporous insulation on the head loss performance of the newly installed ECCS suction strainers. The tests demonstrated that the head loss for a mixed fibrous insulation/microporous insulation debris bed could be quantified by the same correlations used for other particulate debris components if the microporous to fiber mass ratio does not exceed 20%. Debris generation and strainer head loss analyses were revised to incorporate a new source term which includes some microporous insulation. The analyses were used to identify specific pipe whip restraints for which replacement of microporous insulation would be required to ensure NPSH margin can be maintained under design basis conditions. This work is scheduled for RO 14 under a separate design change package (Ref. 7.47) and is tracked as Action Item 5.1.

Replacement of the microporous insulation during RO 14 will complete actions required to address issues cited in Bulletin 96-03. The design basis for ECCS suction strainers will be updated to reflect criteria documented in design change package F1-98-100. These changes will be documented in the FSAR reflecting completion of the licensing commitments made for resolution of the debris generation and transport concerns raised in the Bulletin.

3.0 SAR REVIEW

3.1 A FOLIO search of the JAF FSAR (Ref. 7.10) and Technical Specifications (Ref. 7.7) was performed using the following words: High Pressure Coolant Injection, HPCI, Reactor Core Isolation Cooling, RCIC, Emergency Core Cooling, ECCS, Suppression Pool, Loss of Coolant, LOCA, Small Break, Intermediate Break. 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, 3.2.4, 4.7, 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. 7.11)
- NEDC-32016P-1, Rev.1, Power Uprate Safety Analysis for the James A FitzPatrick Nuclear Power Plant, prepared by the General Electric Company (Ref. 7.12)
- NSE JAF-SE-96-048, Rev. 2, Revision to FSAR to Raise Maximum Allowable Lake Temperature form 82°F to 85°F (Ref. 7.13)
- NSE JAF-SE-96-052, Rev. 0, FitzPatrick Cycle 13 Core Reload Safety Evaluation (Ref. 7.46)
- Technical Specifications 3.5/4.5, 3.7/4.7, 5.0, including the Bases
- Core Operating Limits Report, Rev. 5

3.2 The following sections of the FSAR were revised following RO 13 to incorporate the modification:

- 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 The following sections of the FSAR will require revision following RO 14 to incorporate changes to the plant design and licensing bases for the ECCS suction strainers:

- Section 6.4.1 addresses HPCI NPSH as follows: "To ensure positive suction head to the pump, it is located below the water levels in the suppression pool and condensate storage tanks. The NPSH requirement is met since the available NPSH is 43 ft while only 16 ft is required". This statement should be revised to reflect the impact of small break LOCA conditions (suppression chamber absolute and vapor pressures) and strainer head loss associated with debris generation and transport to the pool.
- Section 6.5.1 currently describes the factors considered in calculating NPSH for the ECCS pumps. This discussion should be revised to include head loss across suction strainers due LOCA-generated debris blockage.

3.4 No changes to the Technical Specifications are required.

4.0 REVIEW AND ANALYSIS

The HPCI System is one of the Emergency Core Cooling Systems (ECCS). It provides reactor vessel inventory makeup during small and intermediate break loss-of-coolant accidents (LOCA) to prevent fuel clad melting. The HPCI System permits the plant to be shut down while maintaining sufficient reactor vessel water inventory until the reactor vessel pressure is below the value at which either the LPCI or Core Spray System can maintain core cooling. If a LOCA occurs, the reactor scrams upon receipt of a low water level signal from the reactor or a high pressure signal from the drywell. The HPCI System controls automatically start the system and bring it to design flow rate within 30 seconds from receipt of a reactor vessel low-low water level signal or a drywell high pressure signal. The HPCI System automatically stops when it receives a signal of high water level in the reactor vessel.

HPCI provides its rated flow over a reactor pressure range from 150 psig to a maximum pressure based on the lowest SRV safety setpoint. The system has five modes of operation. Emergency cooling is the only mode where suction from the suppression pool may be used. The HPCI system is maintained in a standby status during normal operation.

The RCIC System is designed to provide makeup to the Reactor Coolant System (RCS) during isolation of the reactor vessel from the main condenser. Though not part of the ECCS, RCIC also serves as a redundant makeup system in the event of a total loss of all offsite power if HPCI is unavailable. The RCIC System provides core cooling during reactor isolation by pumping makeup water into the reactor vessel in response to low vessel water level. The system is capable of supplying sufficient makeup water so that actuation of ECCS is not required for events other than pipe breaks or loss of reactor coolant inventory. The system is also designed to operate with the RHR System in the steam condensing mode.

Both HPCI and RCIC systems are normally aligned to take suction from the Condensate Storage Tanks (CST). One hundred thousand gallons in each of the two tanks is reserved for the use of the HPCI and RCIC systems. Should the CST be drawn down to a low level, HPCI and RCIC suction is automatically transferred to the suppression pool. This transfer can also be manually initiated. Water from each system is pumped into the reactor vessel via separate feedwater lines.

All components necessary to initiate operation of the HPCI and RCIC Systems are completely independent of emergency AC power, requiring only DC power from the Plant Battery System. The power source for the turbine drivers is the steam generated in the reactor vessel by decay heat in the core. The steam is piped directly to the pump turbine drivers; turbine exhaust is piped to the suppression pool.

4.1 Debris Generation and Transport

The maximum quantities of debris that might be destroyed and transported to the suppression pool was estimated in DE&S calculation A384.F02-02 (Ref. 7.14) using potential break locations identified for the large break analysis and methodology developed in the URG (Ref. 7.5). The analysis, developed for evaluation of RHR and Core Spray suction strainers, considered debris sources from fibrous piping insulation, metallic foil (RMI) from insulation installed on the reactor vessel, suppression pool sludge, particulate debris (e.g. paint chips, dirt, rust) and other miscellaneous debris. The evaluation examined potential break locations in high energy piping 12" and larger within the drywell, the shape and size of the Zone of Influence (ZOI) for each type of insulation material and the quantity of insulation located within the ZOI.

HPCI (and potentially RCIC) operation at full flow could be required for an extended period of time for small breaks up to 0.1 ft^2. For intermediate breaks up to 0.2 ft^2, HPCI could be required for up to 10 minutes. The operational requirements for HPCI correspond to equivalent pipe break sizes of 4.28" (0.1 ft^2) and 6.06" (0.2 ft^2). Design input data was not readily available for high pressure piping in the drywell with diameters in this range. To evaluate small and medium break conditions, the analysis assumed 4.28" and 6.06" diameter breaks for each large break weld location. A double-ended guillotine zone of influence (ZOI) was used for conservatism. A relevant destruction and transport factor was then applied to estimate the amount of debris that would be transported to the suppression pool.

The analysis was revised following design and installation of the strainers during RO 13 to incorporate the new source term with remaining microporous insulation.

4.2 Net Positive Suction Head (NPSH) Evaluation used in Strainer Design

The original General Electric Design Specifications for the RHR and Core Spray system pumps required a suppression pool suction strainer design that would ensure minimum required NPSH could be maintained with the strainer 50% plugged (Ref. 7.15, 7.16). The Design Specifications for HPCI (Ref. 7.18, 7.19) and RCIC (Ref. 7.20) provided no criteria for strainer blockage. Since the original design basis documentation for HPCI and RCIC did not explicitly define a blockage requirement, the 50% requirement for RHR and Core Spray was used as the current basis for HPCI and RCIC strainer design.

NPSH for HPCI and RCIC was evaluated in Proto-Power calculation 98-019 (Ref. 7.21) assuming a 50% reduction in strainer surface area, to account for debris loading consistent with the original design basis, and for design debris loading in accordance with the criteria of Reg. Guide 1.82, Revision 2. Debris head loss values were determined in DE&S calculation A384.F02-06 (Ref. 7.22) based on correlations presented in NUREG/CR-6224 and analysis of testing performed for the PCI stacked disk strainers at the EPRI strainer test facility.

Proto-Power calculation 98-019 evaluated HPCI NPSH margin at long-term conditions for a small break LOCA. Since RCIC is not credited in the response to any LOCA, NPSH evaluation determined the limiting quantity of fibrous debris in the suppression pool that the strainer could accommodate without exceeding the allowable head loss specified in the Procurement Specification (Ref. 7.6). The Specification restricts head loss across strainer disks to 5 feet maximum to limit structural loading on the perforated plate. The analysis performed for HPCI and RCIC verified that adequate NPSH would be available under all operating conditions described in the safety analysis report with postulated debris deposition.

The current design basis for NPSH analysis differs from that in the FSAR in that the current analysis is based on plant operation at an uprated reactor power level of 2536 MWt (Ref. 7.12) and an increased lake temperature of 85°F (Ref. 7.13). FSAR Figure 6.4-1 specifies HPCI suction flow from the suppression pool during an accident (reactor at high or low pressure) as 4250 gpm at 140°F; FSAR Figure 4.7-3 specifies RCIC suction flow from the suppression pool during an accident (reactor at high or low pressure) as 400 gpm at 140°F (Ref. 7.10). A December 1990 letter from GE to NYPA (Ref. 7.23) indicates that there could be a 4°F suppression pool temperature increase, associated with power uprate operation, for a small break LOCA or during reactor isolation. The GE containment response analysis for increased ultimate heat sink (Ref. 7.24) stated that the RHR Service Water temperature increase, associated with the increase in ultimate heat sink temperature, may result in a corresponding increase in the suppression pool water temperature following a transient or accident event due to the reduction in heat removal capability of the RHR heat exchangers. For an increase in lake temperature from 82°F to 85°F, therefore, suppression pool temperature could increase by as much as 3°F.

The NPSH analysis (Ref. 7.21) considered the cumulative impact of power uprate and ultimate heat sink design changes in its determination of a design basis suppression pool temperature. The impact of these two design changes could result in an increase in peak suppression pool temperature for HPCI and RCIC design cases from 140°F to 147°F. This value was conservatively rounded to 150°F for strainer design.

Long-term hot standby operation of the HPCI and RCIC pumps without AC power is discussed in FSAR Section 4.7.6. The RCIC and HPCI Systems are designed for startup and short-term operation without AC power. However, AC power is required for suppression pool cooling (via RHR System) to limit pool temperature for long-term operation. If suppression pool cooling is not available, RCIC and HPCI operation would be limited based on available NPSH. Without the RHR System heat removal capability, the suppression pool would reach 188°F in about 5 hr (Ref. 7.10). The NPSH analysis in Ref. 7.21 shows adequate NPSH margin at 188°F for both systems with no debris deposition. Since complete loss of AC power to essential buses concurrent with a small break LOCA is not within the plant design basis, the analysis confirms HPCI and RCIC operation is not adversely impacted by this design change for events considered in FSAR Section 4.7.6.

Conditions evaluated in the NPSH analysis were specified in Procurement Specification JAF-SPEC-MISC-02871 (Ref. 7.6). The conditions analyzed are summarized in Table 4.1.

TABLE 4.1
NPSH EVALUATION SYSTEM CONFIGURATIONS

SYSTEM	TIME (sec)	FLOW (gpm)	SUPPRESSION POOL TEMP (°F)
HPCI	> 600	4250	150
RCIC	> 600	400	150

4.3 Containment Overpressure

Consistent with the original plant licensing basis (Ref. 7.11), the HPCI and RCIC systems were evaluated at standard atmospheric pressure, i.e. no credit was assumed for containment overpressure.

4.4 Strainer Hole Sizing

No strainer criteria were given in the original General Electric Design Specifications for HPCI or RCIC (Refs. 7.18, 7.19, 7.20). GE Service Information Letter No. 323 (Ref. 7.25) 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 disk strainer assemblies uses perforated plate with 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.

FSAR Section 3.2.4 discusses the smaller lower tie plate (LTP) flow holes designed into the newer fuel assemblies in use at JAF. The flow hole sizes vary, ranging down to 0.228" diameter for Atrium-10A fuel (Ref. 7.46). The smaller hole sizes are intended to provide increased protection against the introduction of debris into the fuel bundles. The LTP hole sizes are larger than the 3/32" diameters holes in the new ECCS suppression pool suction strainers. This ensures that debris small enough to pass through the torus strainers would not be capable of blocking fuel assembly flow.

4.5 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 potential for air/nitrogen ingestion into the HPCI and RCIC strainers as a result of bubbles discharged from the SRV T-Quenchers and the LOCA downcomers was evaluated in DE&S calculation A384.F02-04 (Ref. 7.26).

During a large break LOCA, the initial bubble from the downcomers (predominantly nitrogen) is discharged within a fraction of a second as a result of the rapid pressure increase in the drywell. Depending upon the proximity of a pump suction strainer to one of the downcomers, there is a potential for the large bubble to envelop the strainer and, if the pump is in operation, for ingestion into the system. LOCA downcomer bubble sizes have not been quantified for small or intermediate break events. Since the pressure increase from a break <0.2 ft^2 would be significantly less than that for the design basis accident, it can be concluded that the pressure differential acting on the 96 downcomers would not be large enough to force a large bubble into an adjacent strainer. The bubbles discharged through the downcomers as a result of the small/intermediate break will be small in nature. Insufficient driving force would exist to displace downcomer bubbles towards the strainers and buoyancy would cause any small bubbles generated to rise to the pool surface. HPCI and RCIC pump performance, therefore, is not expected to be affected adversely by air ingestion from the downcomers.

Following Safety Relief Valve (SRV) actuation, the compressed air/nitrogen in the discharge line is discharged into the suppression pool forming a high pressure bubble. Depending upon its proximity to the T-Quenchers, there is a potential for the air bubble to overlap a suction strainer. The possibility of air ingestion into the HPCI pump from SRV discharge was not considered credible because of the angle of the T-Quenchers and the location of the HPCI strainer between the ends of two T-Quenchers. Similarly, air ingestion through the RCIC strainer is not postulated due to the location of the strainer and the corresponding T-Quencher which are located at opposite ends of the torus bay.

4.6 Vortex Limits

The minimum suppression pool level to prevent vortexing is not specified in the EOP's for HPCI or RCIC pump suction. An evaluation was performed in DE&S calculation A384.F02-03 (Ref. 7.27) to ensure that the minimum submergence is provided with the pool at its Technical Specification minimum level, 13.88' (Ref. 7.7). The analysis used 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 shows the submergence needed to prevent vortexing is dependent on flow as the square of the entrance velocity. Using a maximum flow of 4250 gpm for the HPCI strainer and 400 gpm for RCIC strainer, the analysis shows that the minimum pool level required to preclude vortexing is 10.3' for HPCI and 5.1' for RCIC.

The torus nozzles for HPCI and RCIC are angled up from the penetration centerlines. Although the strainer centerlines are not parallel with the water level, the flow patterns generated by the strainers are comparable to a strainer with its centerline parallel to the water level. The elevation of the centerline at the end of the strainers (closest to the pool surface) was used conservatively as the submergence of the core tube.

4.7 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. 7.24). 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. 7.28). The requirement in NUREG-0783 is intended to prevent unstable condensation at the SRV discharge into the suppression pool. Replacement of the HPCI and RCIC suction strainers does 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.

4.8 Structural Analysis and Qualification

Since the installation of the new strainers involves adding 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 Plant Unique Analysis Report (PUAR) (Ref. 7.29, 7.30) to the extent practicable. Techniques used which differ from the original plant unique analysis were summarized in a supplement to the FitzPatrick PUAR (Ref. 7.31).

The replacement strainers 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 HPCI and RCIC suction strainers followed the generic requirements of NUREG-0661 (Ref. 7.32) and the plant unique methodologies as specified in the PUAR. In addition to deadweight, thermal and seismic loads, the strainers 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 F1-98-100 (Ref. 7.1).

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. A parametric study was performed to evaluate the impact of the replacement strainers on external torus attached piping (Ref. 7.33). The study determined that replacement of the HPCI strainer would result in a significant increase in mass and submerged structure loads, and more detailed analysis was performed. Re-analysis of the HPCI torus penetration, X-226, internal piping up to and including the new suction strainer, and external piping up to the nozzle on Booster Pump 23P-1B was performed in DE&S calculation A384.F02-13 (Ref. 7.34). The analysis demonstrated that, with modification of existing support (PFSK-2305) downstream of 23MOV-58 and addition of a new vertical support (PFSK-9169) upstream of 23MOV-57, piping stresses, penetration loads, and pump/valve nozzle loads will be within applicable ASME code limits.

The new RCIC strainer is shorter but wider than the original strainer. The parametric analysis showed that the new RCIC strainer configuration will not adversely affect the dynamic characteristics of the existing model and will not result in any additional load being transferred to the external piping. Although axial loads on the penetration increased with the new strainer, the existing pool swell load combinations are bounded by the load included in the existing analysis. The penetration qualification, therefore, remains acceptable and within code limits. The parametric study concluded the replacement will not impact the existing analysis, and the torus attached piping and support analysis for RCIC remain valid. No further analysis was required.

4.9 Hydrodynamic Loads

Mark I hydrodynamic submerged structure loads for the ECCS and RCIC suppression pool strainers were calculated in DE&S calculation A384.F02-07 (Ref. 7.35). The development of the submerged structure loads on the HPCI and RCIC strainers followed the requirements of the Load Definition Report (LDR) (Ref. 7.36), NUREG-0661 and the PUAR except for the use of reduced acceleration drag volumes to account for the geometry and perforated nature of the stacked disk strainer assemblies. The original plant unique analysis, the LDR and the Mark I Application Guides did not provide guidance for modeling perforated structures.

Acceleration drag loads on submerged structures are proportional to the acceleration drag volume of the structure. The theoretical acceleration drag volume for an infinitely long solid cylinder is 2.0 times the displaced water volume ($2\pi R^2L$); thus the effective hydrodynamic mass coefficient C_m equals 2.

Acceleration drag tests were performed on the PCI prototype strainer assemblies by DE&S to establish the total inertial mass that will act on the strainer when vibrating in water. The test results (Ref. 7.37, 7.38) were used to calculate a the reduction in load due the holes in the strainer. Conservatively, the total added water mass was found to be 50% of the values obtained for the same shape without holes. The 50% reduction was based on a tested strainer that had $^1/_8$" diameter holes and 40% open area.

The geometry of the FitzPatrick strainers, including stacked disk width and gap width, is consistent with the tested strainers, except that the Fitzpatrick strainers have only 33% open area. Linear interpolation of the test data was used to approximate reduction in the acceleration drag volume. For the Fitzpatrick strainers with 33% open area, interpolation yielded a 40% reduction. Conservatively only a 25% reduction in loads due to the effect of perforations was used for the RHR and Core Spray strainers (reference F1-97-031). Analysis for the HPCI and RCIC strainers used the 40% reduction based on the test results.

In addition to strainer perforations, a reduction in load to account for the finite length of the strainers was developed for HPCI and RCIC. Since the HPCI and RCIC strainers are relatively short, the end effects term becomes significant. The reduction factors, based on the length and diameter of each strainer, were conservatively calculated as 0.915 and 0.705 for the HPCI and RCIC strainers, respectively. Combining the two reduction factors, the effective hydrodynamic mass coefficients (C_m) used in the piping analyses were 1.10 for HPCI and 0.85 for RCIC.

Subsequent to the development of the C_m used in the ECCS and RCIC piping analyses, additional strainer hydrodynamic mass testing (Ref. 7.39) showed that the values used were conservative. Empirical equations derived from the test results showed that C_m is < 0.5 for these geometries. Supplemental analysis of the HPCI system was performed using a C_m = 0.75 to reduce excess conservatism and more realistically determine the accelerations of valve 23MOV-58.

Analysis of fluid structure interaction (FSI) loads, documented in the existing PUAR, utilized accelerations calculated by Continuum Dynamic, Inc. (CDI) using the PUAR torus shell accelerations and included the direct Condensation Oscillation and Chugging loads on submerged structures. The 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. The FSI loads were applied as force time histories and the results were combined absolutely with the direct submerged structure loads and the torus motion loads. The FSI methodology changes were documented in the PUAR Supplement (Ref. 7.31).

4.10 Load Combinations

In the structural analysis for the new HPCI strainer, independent dynamic loads were combined using the Square Root Sum of the Squares (SRSS) method. This method was used in the evaluation of certain 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. 7.40) 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 non-exceedance probability of 84%. NEDE-24632 used the CDF methods to show that the SRSS combination of independent Mark I dynamic loads has a non-exceedance 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. 7.41). This method has been widely used throughout the industry on Mark I plants. A supplement to the PUAR documented the use of SRSS for FitzPatrick (Ref. 7.31).

4.11 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. 7.42), 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 was documented in a supplement to the PUAR (Ref. 7.31).

4.12 HPCI Pump Suction Valve

The accelerations acting on HPCI pump suction isolation valve, 23MOV-58, were predicted to increase significantly as a result of the strainer replacement (Ref. 7.34). The valve has been evaluated to ensure structural integrity would be maintained under all design conditions (Ref. 7.43). The accelerations acting on the motor operator are bounded by the Limitorque qualification (Ref. 7.44).

4.13 Dose Rates

It is expected that the strainers will accumulate radioactive particulates during normal plant operation in addition to the deposition of debris during an accident. Since the strainers will not generate any additional particulates, offsite dose rates will not be affected. Local dose rates within the torus and external to the torus in the area of the strainers may be impacted during normal plant activities. These would be evaluated and controlled as necessary in accordance with plant procedures to minimize personnel exposure. In an accident scenario, water from the suppression pool is circulated through the systems increasing dose rates in the areas of the pumps. For accident response, it is expected that the dose rate in any area in which operator actions are required would not be affected significantly.

4.14 10CFR 50.59 Evaluation

Replacement of the suppression pool suction strainers for the HPCI and RCIC pumps and changes to plant design and licensing bases to incorporate debris generation and deposition on the HPCI and RCIC suction strainers:

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

   HPCI and RCIC 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 the HPCI pump results in a decrease in moderator temperature. Since this modification involves only the installation of passive pump suction strainers, there is no impact on any pump control function and this modification can not increase the probability of an accidental pump start.

   The replacement of the suppression pool strainers for these systems with the new strainer assemblies, therefore, does 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.**

   HPCI is designed to provide reactor vessel inventory makeup during small and intermediate break loss-of-coolant accidents (LOCA) to prevent fuel clad melting. System operation is independent of AC power. RCIC serves as a redundant makeup system in the event of a total loss of all offsite power if HPCI is unavailable. Installation of new passive suction strainers will not affect the operation or performance of either system.

Analysis performed for the HPCI and RCIC system pumps verified that, with a 50% reduction in suction strainer surface area or with limiting debris deposition on the strainers, adequate Net Positive Suction Head (NPSH) would be available under all operating conditions described in the safety analysis report. The replacement strainers have larger surface areas than those currently installed, increasing the likelihood of long-term pump availability following a small or intermediate break LOCA.

In the event of an accident, the 3/32" hole sizes in the replacement strainers will provide improved protection against blockage of lower tie plate flow holes and introduction of debris into the fuel bundles. The possibility of fuel cladding damage, therefore, is reduced by the modification.

Thermal input to the suppression pool is not increased and the heat capacity of the RHR heat exchangers is not affected. The reduction in suppression pool inventory resulting from installation of the larger HPCI and RCIC strainers is negligible. The suppression pool level specified in Technical Specification 3.7 and minimum pool water volume described in Technical Specification Bases 3.7 are not affected. Suppression pool temperatures under accident conditions, therefore, are not affected by the modification.

The strainers, torus penetration, 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.

Dose rates in any area in which operator actions are required during an accident would not be affected significantly by the debris accumulation on the strainers. Dose rates in the areas around the HPCI and RCIC pumps increase as a result of circulating suppression pool water through the systems when the pumps are in operation. The additional source term associated with debris accumulated on the strainers would not add to the dose rates in those areas substantially.

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

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 greater than those of the original strainers to accommodate a larger accumulation of debris, while still meeting pump net positive suction head requirements. Strainer 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. 7.21) verifies that, with 50% reduction in strainer surface area or with the limiting debris mix deposited on the strainers, the HPCI and RCIC pumps would have adequate $NPSH_{AVAILABLE}$ to perform their accident mitigation functions. The perforated holes in the disk assemblies were sized to ensure particles do not pass through which could plug system clearances.

The evaluations in Ref. 7.26 conclude that, for HPCI and RCIC, vapor bubble ingestion as a result of LOCA downcomer discharge or SRV initiation is not credible for a small or intermediate break accident.

The new strainer assemblies were structurally evaluated to ensure integrity with debris loading. The replacement suction strainers, torus attached piping and torus penetrations 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 provide the same function as the original 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.

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. 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.

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 original 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 were the original strainers. No new failure modes or system interactions are introduced by the modification and independence of the safety trains is not affected.

   HPCI and RCIC system control functions and independence from AC power are not impacted in any way by this modification. The ability of either system to provide core cooling following a loss of AC power is unaffected.

   The evaluations in Ref. 7.26 conclude that vapor bubble ingestion, as a result of LOCA downcomer discharge or SRV initiation, is not credible for a small or intermediate break accident.

   The materials of the strainer assemblies are fully compatible with the Reactor Coolant System and support the structural requirements of the design. The new strainers and piping 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 have been 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 function of the HPCI system as providing adequate cooling to the core for all pipe breaks smaller than those for which the LPCI or Core Spray systems can protect the core. RCIC is described as a redundant makeup system for total loss of all offsite power in the event HPCI is unavailable. Installation of passive suction strainers in the suppression pool will not affect system performance or impact, in any way, the initiating functions for either pump.

   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 reduction in suppression pool inventory resulting from installation of the larger HPCI and RCIC strainers is negligible. The minimum pool water volume described in Technical Specification Bases 3.7, therefore, remains unchanged and the water level required to ensure complete condensation during an accident is maintained. Also, the suppression pool water volume required to act as the heat sink for the RCS energy release following a postulated LOCA has not been 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.

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 were procured and installed in accordance with the applicable requirements for Category I components. There has been no impact on the Quality Assurance Program.

The changes did not impact the site fire protection program. No fire barrier was affected, no flammable or combustible material was added or removed from any area of the plant, and no fire detection/suppression equipment or emergency lighting was impacted. FPES-04A Exhibit 1, (Ref. 7.45) 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, was controlled by existing plant procedures which specify use of pre-approved paints. The modification had no other effect on air or water quality, did not involve use of any hazardous substance, did not affect any property outside the plant buildings, did not introduce any new effluent paths, involve storage of any radioactive material, or affect the Meteorological Tower.

The installation of replacement strainers for the HPCI and RCIC pumps did not affect the operation or performance of any component or system required for accident mitigation. No EOP revisions were required. This modification had no impact on Emergency Plan staffing, notification, monitoring or reporting systems. No structures outside the reactor building were affected.

5.0 ACTION ITEMS TO BE TRACKED

5.1 Removal of microporous insulation from pipe whip restraints in the drywell at locations delineated in the debris generation and transport analysis, DE&S calculation A384.F02-02, must be completed to validate the assumptions of the Proto-Power Corp. Calculation No. 98-019, Rev. D. The replacement of microporous insulation for the specified restraints is included in the scope of Minor Modification M1-97-131 and is scheduled for implementation during RO 14.

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

Modification F1-98-100, Rev. 0, "High Pressure Core Injection and Reactor Core Isolation Cooling Suppression Pool Suction Strainer Replacement", installed new suppression pool suction strainers for HPCI and RCIC system pumps. The replacement strainers are larger than the original strainers and provide additional margin for debris loading following a small or intermediate break loss-of-coolant-accident. The strainer assemblies have been analyzed and qualified for seismic and Mark I hydrodynamic loads specified in the FSAR and Plant Unique Analysis Report.

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 has not been 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 F1-98-100, Rev. 0, High Pressure Coolant Injection and Reactor Core Isolation Cooling Suppression Pool Suction Strainer Replacement

7.2 USNRC Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors, May 6, 1996

7.3 NUREG/CR-6224, Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris, October 1995

7.4 Regulatory Guide 1.82, Rev. 2, Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident, May 1996

7.5 NEDO-32686, Draft, Utility Resolution Guidance for ECCS Suction Strainer Blockage, prepared by the Boiling Water Reactor Owners' Group, November 1996

7.6 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

7.7 Technical Specifications 3.5/4.5, 3.7/4.7, Updated through Amendment 240

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

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

7.10 FSAR, 1997 FSAR Update

7.11 U.S. Atomic Energy Commission Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant, Docket No. 50-333, November 20, 1972

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

7.13 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

7.14 Duke Engineering & Services Calculation No. A384.F02-02, Rev. 4, JAF Nuclear Plant: Estimation of Debris Generation and Transport to the Suppression Pool following a LOCA

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

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

7.17 Michael J. Colomb, JAF Site Executive Officer, letter to USNRC, JAFP-98-0306, Information Regarding NRC Bulletin 96-03, September 16, 1998

7.18 General Electric Design Specification 22A1362, Rev. 5, High Pressure Coolant Injection System

7.19 General Electric Design Specification 22A4313, Rev. 0, High Pressure Coolant Injection System

7.20 General Electric Design Specification 22A4314, Rev. 0, Reactor Core Isolation Cooling System

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

7.22 Duke Engineering & Services Calculation No. A384.F02-06, Rev. 2, Strainer Head Loss Performance Analysis (ITS)

7.23 C.A. Stool, GE Nuclear Energy, letter to Richard Chi, New York Power Authority, JA FITZPATRICK (JAFNPP) POWER UPRATE PROGRAM - Assessment of Impact on HPCI and RCIC Systems, 12/21/90

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

7.25 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

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

7.27 DE&S Calculation No. A384.F02-03, Rev. 1, RHR, CS, HPCI and RCIC Suction Strainer Vortex/Minimum Submergence

7.28 NUREG-0783, Suppression Pool Temperature Limits for BWR Containments, November 1981

7.29 Teledyne Engineering Services, Technical Report TR-5321-1, Mark I Containment Program Plant Unique Analysis Report of the Torus Suppression Chamber for James A. Fitzpatrick Nuclear Power Plant, Rev. 1, November 1984

7.30 Teledyne Engineering Services, Technical Report TR-5321-2, Mark I Containment Program Plant Unique Analysis Report of the Torus Attached Piping for James A. Fitzpatrick Nuclear Power Plant, Rev. 1, November 1984

7.31 JAF-RPT-MULTI-03000, Rev. 1, ECCS Suction Strainer Replacement Modification Supplement to the Plant Unique Analysis Report

7.32 NUREG-0661, Safety Evaluation Report Mark I Containment Long-Term Program, Resolution of Generic Technical Activity A-7, July 1980, including Supplement 1, August 1982

7.33 JAF-RPT-MULTI-02921, Rev. 1, Torus Attached Piping Parametric Study for the ECCS Suction Strainer Replacements

7.34 DE&S Calculation No. A384.F02-13, Rev. 0, HPCI Penetration X-226 TAP Piping Reanalysis for the Replacement Suction Strainer Assemblies

7.35 DE&S Calculation No. A384.F02-07, Rev. 1, Mark I Hydrodynamic Submerged Structure Loads on the Replacement Core Spray, RHR, HPCI, and RCIC Suction Strainer Assemblies

7.36 NEDO-21888, Rev. 2, November 1988, General Electric, Mark I Containment Program Load Definition Report

7.37 DE&S Test Report No. TR-ECCS-GEN-01-NP, Rev. 2, Test Report for Hydrodynamic Inertial Mass Testing of ECCS Suction Strainers

7.38 DE&S Test Report No. TR-ECCS-GEN-05-NP, Rev. 1, Supplement 1 to Hydrodynamic Inertial Mass Testing of ECCS Suction Strainers - Free Vibration Analysis

7.39 DE&S Test Report No. TR-ECCS-GEN-011, Rev. 0, ECCS Suction Strainer Hydrodynamic Test Summary Report

7.40 GE Report NEDE-24632, Mark I Containment Program - Cumulative Distribution Functions for Typical Dynamic Responses of a Mark I Torus and Attached Piping Systems, December 1980

7.41 D.B. Vassallo (USNRC) letter to H. C. Pfefferlen (GE), Acceptability of SRSS Method for Combining Dynamic Responses in Mark I Piping Systems, March 10, 1983

7.42 NEDO-24583-1, "Mark I Program, Structural Acceptance Criteria, Plant Unique Analysis Application Guide," October 1979

7.43 Altran Calculation No. 93179-C-61, Rev. 0, Valve Thrust Assessment 16" Powell Gate Valve 23MOV-58

7.44 Limitorque Report B0115, Hydrodynamic Vibration Test (New Loads), 6/24/82

7.45 ESM: FPES-04A, Rev. 1, Fire Protection/Appendix R Compliance Procedure

7.46 JAF-SE-96-052, Rev. 0, FitzPatrick Cycle 13 Core Reload Safety Evaluation

7.47 JAF Minor Modification M1-97-131, Rev. 0, Drywell Pipe Mineral Wool Insulation Replacement and Drywell Pipe Rupture (Whip) Restraint Insulation Replacement