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JAN 1 8 2003

PSEG
Nuclear LLC

LR-N02-0417
LCR H02-01

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Gentlemen:

HOPE CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION REGARDING RELAXATION OF SECONDARY CONTAINMENT OPERABILITY REQUIREMENTS AND ELIMINATION OF FRVS RECIRCULATION CHARCOAL FILTERS
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Reference: Letter LR-N02-0002, *Request For Change To Technical Specifications Relaxation Of Secondary Containment Operability Requirements And Elimination Of FRVS Recirculation Charcoal Filters*, dated June 28, 2002

On June 28, 2002 PSEG Nuclear LLC (PSEG) submitted the referenced request for a revision to the Technical Specifications (TS) to request relaxation of Secondary Containment Operability Requirements and to eliminate the Filtration, Recirculation and Ventilation System (FRVS) charcoal filters for Hope Creek Generating Station. Mr. George Wunder, NRC Hope Creek Project Manager, advised PSEG that the NRC could not approve the request unless it was in-line with the guidance of TSTF-51, Rev. 2, specifically the use of the phrase "recently irradiated fuel." PSEG agreed to modify the Licensing Change Request to be consistent with TSTF-51, Rev. 2 . Attachments 1 and 2 of our June 28, 2002 letter have been revised and are included with this letter. Changes are noted by marginal markings. There has been no change to the No Significant Hazards Analysis. Attachments 3 and 4 to our June 28, 2002 letter do not require revision to support the request.

If you have any questions or require additional information, please contact Mr. Michael Mosier at (856) 339-5434.

A001

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 1/18/03

Sincerely,

D.F. Garchow

D. F. Garchow
Vice President-Projects and Licensing

Attachments

JAN 1 8 2003

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS FOR RELAXATION OF SECONDARY CONTAINMENT REQUIREMENTS AND ELIMINATION OF FRVS RECIRCULATION CHARCOAL FILTERS

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS RELAXATION OF SECONDARY CONTAINMENT OPERABILITY REQUIREMENTS AND ELIMINATION OF FRVS RECIRCULATION CHARCOAL FILTERS

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS RELAXATION OF SECONDARY CONTAINMENT OPERABILITY REQUIREMENTS AND ELIMINATION OF FRVS CHARCOAL FILTERS

1. DESCRIPTION

The purpose of this request for changes to the Technical Specifications (TS) is to provide flexibility in scheduling outage tasks and to modify unnecessarily restrictive containment closure and ventilation system requirements. The elimination of the selected TS Engineered Safety Feature (ESF) requirements during core alterations and the movement of sufficiently decayed irradiated fuel is consistent with TSTF-51, Revision 2 to NUREG-1433 Vol. 1, Rev. 2, Standard Technical Specifications, General Electric Plants. Analysis of the Fuel Handling Accident utilizing Alternate Source Term (AST) has been performed to support this change. Also, the Loss of Coolant Accident (LOCA) analysis has been redone to support the elimination of the Filtration, Recirculation and Ventilation System (FRVS) Charcoal Filters.

The definition of CORE ALTERATION will be revised to exclude control rod movement, provided there are no fuel assemblies in the associated core cell, as a CORE ALTERATION. In addition, the limiting condition for operation (LCO) of the FRVS ventilation is revised to allow operation beyond seven days as long as the other unit is placed into operation. This is consistent with Standard Technical Specifications (STS) NUREG-1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

The Fuel Handling Accident (FHA) has been analyzed using a AST (10CFR50.67). This allows relaxation of Secondary Containment OPERABILITY requirements. The proposed amendment would revise the Hope Creek TS in Appendix A of the Operating License to eliminate the requirement for secondary containment when handling irradiated fuel and during CORE ALTERATIONS. Operability would be required, however, for handling recently irradiated fuel and for operations with the potential for draining the reactor vessel. The analysis does not take credit for the Control Room Emergency Ventilation (CREF) System.

The LOCA dose calculation has been revised to (1) eliminate credit for the FRVS recirculation charcoal filters, (2) reduce credited efficiency of FRVS vent charcoal filters by eliminating the heaters, (3) reduce maximum permitted ESF leakage from 10 gpm to 1 gpm and (4) reduce control room unfiltered in-leakage to 350 cfm based upon testing results. The loss of coolant accident was analyzed using an alternate source term (AST) methodology to support deletion of the main steam line leak detection system during RFO10. This was issued on October 3, 2001 as Amendment No. 134.

2. PROPOSED CHANGES

The proposed revisions to the HCGS TS are being made to implement AST, as described in Regulatory Guide (RG) 1.183, through reanalysis of the radiological consequences of a FHA. This will allow relaxation of secondary containment operability requirements during refueling. The elimination of the selected TS ESF requirements during core alterations and the movement of sufficiently decayed irradiated fuel is consistent with TSTF-51, Revision 2 to NUREG-1433 Vol. 1, Rev. 2, Standard Technical Specifications, General Electric Plants. As part of this implementation, the total effective dose equivalent (TEDE) acceptance criteria of 10CFR50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11 and General Design Criteria (GDC) 19 of 10CFR50, Appendix A. The Bases for TS 3/4.6.5, 'Secondary Containment', is also being changed to maintain the defense in depth philosophy. Also, the LOCA analysis has been redone to support the elimination of the FRVS Charcoal Filters.

Specifically the following Sections of the TS are being changed:

1. Definitions, 1.7, "Core Alterations"
2. 3/4 3.2, "Isolation Instrumentation"
3. 3/4 6.5.1, "Secondary Containment Integrity"
4. 3/4 6.5.2, "Secondary Automatic Containment Isolation Dampers"
5. 3/4 6.5.3, "Filtration, Recirculation and Ventilation System (FRVS)"
6. 3/4 7.1.1, "Safety Auxiliaries Cooling System"
7. 3/4 7.1.2, "Station Service Water System"
8. 3/4 7.1.3, "Ultimate Heat Sink"
9. 3/4 7.2, "Control Room Emergency Filtration System"
10. 3/4 8.1.2, "Electrical Power Systems, A. C. Sources – Shutdown"
11. 3/4 8.2.2, "Electrical Power Systems, D. C. Sources – Shutdown"
12. 3/4 8.3.2, "Electrical Power Systems, Distribution – Shutdown"
13. Bases 3/4 3.2, "Isolation Actuation Instrumentation"
14. Bases 3/4 6.5, "Secondary Containment"

The changes associated with the above Sections of the TS are:

1. Remove "CORE ALTERATIONS"
2. Add "recently" as a modifier to irradiated fuel
3. Modify as appropriate to eliminate the FRVS charcoal filters

Marked up TS pages are included in Attachment 2.

3. BACKGROUND

This TS change is requested to reduce refueling outage critical path time and shorten the overall length of the outage. To move large material into and out of the Reactor Building under the current TS requires that the Reactor Building Truck Bay be used.

The Hope Creek Reactor Building Truck Bay is more cumbersome than most BWR Mark I Containments. Typically, there are two truck or railroad doors, one of which can be kept closed at all times. The Hope Creek truck bay is inside the concrete shield building which is part of secondary containment and includes doors and bolted hatch openings.

Isolation of the refuel floor when the outside truck bay door is open is via the C-9 Hatch which is in the ceiling of the truck bay. The C-9 hatch has been cumbersome to operate and many doors require a certain amount of effort to secure secondary containment when the outside truck bay door is open. Once the outside truck bay door is closed the effort to re-establish air pressure regulation for the door seal also requires a considerable amount of time.

Closing the C-9 hatch to establish secondary containment until the truck bay door can be closed delays critical path work on the refueling floor due to the inability to move items from the truck bay to the refueling floor and the need to disconnect all temporary cables running through the hatch. In order to reduce outage duration, PSEG proposes to relax the requirements for secondary containment based on reanalysis of the FHA using AST.

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed TS changes take advantage of a specific decay period to reduce the radionuclide inventory available for release in the event of an FHA. The specific decay period assumed for HCGS was 24 hours. Following the 24-hour decay period, the primary success path for mitigating the fuel handling accident no longer includes the functioning of the active containment systems. The FHA is the bounding accident during fuel handling and core alterations. Fuel that has not decayed for 24 hours is termed "recently irradiated fuel" and the subject ESF features must remain OPERABLE when moving such fuel. Applying the "recently irradiated fuel" concept to the TS provides a mechanism for defining a minimum time for fission product decay. The decay period of 24 hours has been shown by analysis to provide sufficient decay and is less than the time required to disassemble the reactor pressure vessel and begin moving fuel.

The definition of "CORE ALTERATIONS" is being changed in accordance with TSTF-51, Revision 2. As described in TSTF-51, Revision 2, accidents postulated to occur during core alterations, in addition to fuel handling accidents, are inadvertent criticality (due to control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the proposed TS changes omit the term "CORE ALTERATIONS."

Consistent with TSTF-51, Revision 2, requires licensees incorporating this change to commit to NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," subheading "Containment – Primary (PWR)/ Secondary (BWR)."

The commitment in TSTF-51, Revision 2, was based on a draft version of NUMARC 93-01, Revision 3. When NUMARC 93-01, Revision 3, was approved in July 2000, the guidelines referred to in TSTF-51, Revision 2, were designated as Section 11.3.6.5. Section 11.3.6.5 of NUMARC 93-01 states:

> *Maintenance activities involving the need for open containment should include evaluation of the capability to achieve containment closure in sufficient time to mitigate potential fission product release. This time is dependent on a number of factors, including the decay heat level and the amount of RCS inventory available*
>
> *For BWRS, technical specifications may require secondary containment to be closed under certain conditions, such as during fuel handling and operations with a potential to drain the vessel.*
>
> *In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:*
>
> *During fuel handling and core alterations, ventilation systems and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.*
>
> *A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.*

PSEG will follow the guidelines in Section 11.3.6.5 of NUMARC 93-01, Revision 3, at Hope Creek, during refueling within containment. Plant procedures will be revised, as appropriate, to implement these guidelines. This will also be placed in the Bases for Section 3/4.6.5, Secondary Containment.

The LOCA dose calculation, which is the bounding dose analysis, has been revised to (1) eliminate credit for FRVS recirculation charcoal filters, (2) reduce credited efficiency of FRVS vent charcoal filters, (3) reduce ESF leakage from 10 gpm to 1 gpm, and (4) reduce control room unfiltered in-leakage to 350 cfm based upon testing. The loss of coolant accident was analyzed using an AST methodology to support deletion of the MSIV Sealing System (MSIVSS) that was approved as Amendment No.134.

4. TECHNICAL ANALYSIS

4.1 Fuel Handling Accident (FHA)

The FHA is analyzed using the plant specific design inputs supporting the current licensing bases, which are compatible with the AST and TEDE dose criteria. No specific engineered safety function actuation is credited in the analysis except the scrubbing of the activity in the spent fuel pool, which is limited by 23 feet height of water over the top of irradiated fuel assemblies in the spent fuel pool racks (TS 3/4.9.9, Water Level - Spent Fuel Pool Storage Pool).

The core inventory is calculated based on rated core thermal power. The radial peaking of 1.75 is conservatively used instead of 1.5 recommended in Reference 7.5. The core activity is normalized based on the number of rods failed during the fuel handling accident and core thermal power level to obtain the Ci/MWt. The RADTRAD 3.02 default nuclide inventory file was modified based on the normalized Ci/MWt. The plant-specific nuclide inventory file is further modified to include Kr-131m, Xe-131m, Xe-133m, Xe-135m, and Xe-138 isotopes, which are critical for the early short-term exposure. The RADTRAD 3.02 dose conversion factors were modified to include the dose conversion factors for the added noble gas isotopes. The modified dose conversion factor file is used in the fuel handling accident analysis. Since the dose conversion factors for ground shine dose and dose rate are not used to evaluate the offsite and control room doses, they are set to zero.

The EAB, LPZ, and CR doses are calculated using the post-FHA release through reactor building truck bay door. The activity release rate from the damaged fuel pins is modeled for release of 99 percent of all radioactive material to the environment over a 2-hour period. (Ref. 7.1, Regulatory Position 5.3).

The use of a peaking factor of 1.75 and an activity release rate of 153,533 cfm in the analysis yield conservative radiological consequences at the given receptor locations, but their use also yields high activity concentrations at the control room air intake, which is not conservative with respect to control room air intake monitor response. However, the control room dose is conservatively analyzed with the normal unfiltered air intake flow rate of 3,300 cfm (3,000 cfm + 10 %) for 30 days and CREF initiation is not credited in the FHA analysis. The resultant doses are shown below.

	Fuel Handling Accident Occurring In Reactor Building TEDE Dose (rem)		
	Receptor Location		
	Control room	EAB	LPZ
Calculated Dose	3.21E+00	5.15E-01 (0.0 hr)	5.15E-02
Allowable TEDE Limit	5.00E+00	6.30E+00	6.30E+00

4.2 LOCA Analysis

The LOCA was first analyzed using an AST methodology to support deletion of the MSIV Sealing System (MSIVSS) and approved in Amendment No.134. The resulting doses are shown below.

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (Rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Containment Leakage	4.29E-01	3.41E-01(3.2 hr)	1.10E-01
ESF Leakage	2.64E-01	3.51E-02 (0 hr)	1.19E-02
MSIV Leakage	3.40E+00	1.92E+00 (9.3 hr)	3.67E-01
Containment Purge	0.00E+00	0.00E+00	0.00E+00
Containment Shine	0.00E+00	0.00E+00	0.00E+00
External Cloud	0.00E+00	0.00E+00	0.00E+00
CR Filter Shine	2.46E-03*	0.00E+00	0.00E+00
Total	4.10E+00	2.30E+00	4.89E-01
Allowable TEDE Limit	5.00E+00	2.50E+01	2.50E+01

*CR filter shine dose due to the CR unfiltered inleakage of 1000 cfm with RADTRAD default nuclide inventory file (NIF) will bound that due to the CR unfiltered inleakage of 900 cfm with the plant-specific NIF.

The purpose of Revision 1 to Calculation H-1-ZZ-MDC-1880 is to evaluate the EAB, LPZ, and CR post-LOCA doses for HCGS due to:

- Reduction of CR unfiltered leakage from 900 cfm to 350 cfm. Actual measurements using tracer gas testing was 206 cfm including uncertainty
- Elimination of credit for FRVS recirculation charcoal filters
- Reduction of FRVS vent charcoal filter efficiencies for elemental and organic iodine based on removal of safety grade heaters
- Reduction of ESF leakage from 10 gpm to 1gpm (based on operations)
- Increase core thermal power to be consistent with proposed power uprate project
- Reduction in plate-out fraction based upon current NRC model

The Results of the revised calculation are shown below:

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (Rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Containment Leakage	1.00E+00	4.40E-01(3.4 hr)	1.710E-01
ESF Leakage	1.17E+00	1.79E-02 (14.6 hr)	9.17E-02
MSIV Leakage	1.96E+00	2.45E+00 (9.3 hr)	4.32E-01
Containment Purge	0.00E+00	0.00E+00	0.00E+00
Containment Shine	0.00E+00	0.00E+00	0.00E+00
External Cloud	0.00E+00	0.00E+00	0.00E+00
CR Filter Shine	2.46E-03*	0.00E+00	0.00E+00
Total	4.13E+00	3.07E+00	6.94E-01
Allowable TEDE Limit	5.00E+00	2.50E+01	2.50E+01

*CR filter shine dose is bounding (Calculation H-1-ZZ-MDC-1880, Section 6.4.4, item 5)

Details of the reanalysis are contained in Calculation H-1-ZZ-MDC-1880 Rev. 1IR0, Post-LOCA EAB, LPZ, and CR Doses – Alternate Source Term Analysis (Attachment 4).

4.3 FRVS Ventilation

The TS presently provide a period of 7 days to restore an inoperable FRVS ventilation unit when performing activities with the potential for draining the reactor vessel or discontinue such activities. Equipment alignment and maintenance during refueling outages may result in one FRVS train being inoperable for a period exceeding 7 days. Maintenance activities such as CRDM refurbishment would then be stopped. Operation of the redundant train will ensure that the remaining subsystem is operable, that no failures, which could prevent automatic actuation, have occurred and that any other failures will be readily detected. This is consistent with Standard Technical Specifications (STS) NUREG-1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

5. REGULATORY SAFETY ANALYSIS

5.1. No Significant Hazards Consideration

Amendment No. 134 for Hope Creek Generating Station (HCGS) was issued on October 3, 2001. The changes issued in that amendment were based upon full implementation of an Accident Source Term (AST) pursuant to 10CFR50.67 using the guidance provided in Regulatory Guide (RG) 1.183. Full implementation revises the plant's licensing basis to specify the alternate source term in place of the previous source term and establishes the total effective dose equivalent (TEDE) acceptance criteria in 10CFR50.67 in lieu of the whole body and thyroid dose guidelines provided in 10CFR100.11. Therefore, the licensing basis for the HCGS is alternate source term with TEDE dose criteria.

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

5.1.1. Does the change involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The definition of CORE ALTERATIONS has been revised to define that control rod movement, provided there are no fuel assemblies in the associated core cell, is not a core alteration. This is consistent with Standard Technical Specifications (STS) NUREG-1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

The TS presently provide a period of 7 days to restore an inoperable FRVS ventilation unit when performing activities with the potential for draining the reactor vessel or discontinue such activities. Operation of the redundant train will ensure that the remaining subsystem is operable, that no failures, which could prevent automatic actuation, have occurred and that any other failures will be readily detected. This is consistent with STS, NUREG-1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

The proposed changes associated with the FHA do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report (UFSAR). The FHA for the HCGS is defined as a drop of a fuel assembly over irradiated assemblies in the reactor core 24 hours after reactor shutdown. AST is used to evaluate the dose consequences of a postulated accident. The FHA has been analyzed without credit for Secondary Containment, Filtration Recirculation and Ventilation System (FRVS), and Control Room Emergency Filtration (CREF) system. The resultant radiological consequences are within the acceptance criteria set forth in 10CFR50.67 and Regulatory Guide 1.183. This amendment does not alter the methodology or equipment used directly in fuel handling operations. The equipment hatch, the personnel air locks, nor any other containment penetration, nor any component thereof is an accident initiator. Actual fuel handling operations are not affected by the proposed changes. Therefore, the probability of a Fuel Handling Accident is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

The Loss of Coolant Accident (LOCA) Dose Calculation has been revised to (1) eliminate credit for the FRVS recirculation charcoal filters, (2) reduce credited efficiency of FRVS vent charcoal filters, (3) reduce Engineered Safety Feature (ESF) leakage from 10 gpm to 1 gpm and (4) reduce control room unfiltered in-leakage to 350 cfm. These proposed changes do not eliminate any safety system. The changes are only associated with the credit provided by the system in reducing the radiological consequences and therefore, do not affect any accident initiator. The results of that analysis show that the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses are of the same order of magnitude as the previous analysis and remain within the acceptance criteria in 10CFR50.67 and Regulatory Guide 1.183.

Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.

5.1.2. Does the change create the possibility of a new or different kind of accident from any accident previously analyzed?

Response: No

The proposed amendment will not create the possibility for a new or different type of accident from any accident previously evaluated. Changes to the allowable activity in the primary and secondary systems do not result in changes to the design or operation of these systems. The evaluation of the effects of the proposed changes indicates that all design standard and applicable safety criteria limits are met.

Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes and modifications do not significantly affect the mitigative function of these systems. Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

5.1.3. Does the change involve a significant reduction in the margin of safety?

Response: No

The proposed changes revise the TS to establish operational conditions where specific activities represent situations during which significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the regulatory guidelines. Safety margins and analytical conservatisms are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed TS continue to ensure that the TEDE for the CR, the EAB, and LPZ are below the corresponding acceptance criteria specified in 10CFR50.67 and RG1.183.

Therefore, these changes do not involve a significant reduction in margin of safety.

Based on the above, PSEG concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in

10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2. Applicable Regulatory Requirements/Criteria

Both the FHA and the revision to the LOCA dose analysis were performed using AST and TEDE dose criteria in accordance with the guidance provided in Regulatory Guide 1.183. The RADTRAD V3.02 computer code was utilized to perform the calculations. AST methodology also involves the use of the TEDE criteria provided in 10CFR50.67, which sums dose contributions from inhalation and external exposure. The assumptions and applicable design inputs are documented in Engineering Evaluation H-1-ZZ-MEE-1611and Design Calculation H-1-ZZ-MDC-1880.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7. REFERENCES

7.1. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

7.2. S.L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation", NUREG/control room-6604, USNRC, April 1998.

7.3. 10CFR50.67, "Accident Source Term".

7.4. NEI 99-03 (Draft), February 2001, Control Room Habitability Assessment Guidance.

7.5. U.S. NRC Safety Guide 25, March 23, 1972, Assumptions Used For Evaluating The Potential Radiological Consequences Of A Fuel Handling Accident In The Fuel Handling and Storage Facility For Boiling and Pressurized Water Reactors

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License NPF-57 are affected by this change request:

Technical Specification	Page
1.0	1-2
3/4 3.2	3/4 3-16a
	3/4 3-31
3/4 6.5	3/4 6-47
	3/4 6-49
	3/4 6-51
	3/4 6-51a
	3/4 6-52
	3/4 6-52a
	3/4 6-53
	3/4 6-53a
3/4 7.1	3/4 7-1
	3/4 7-3
	3/4 7-5
3/4 7.2	3/4 7-6
3/4 8.1	3/4 8-11
3/4 8.2	3/4 8-17
3/4 8.3	3/4 8-23
B3/4 3.2	B3/4 3-2g
B3/4 6.5	B3/4 6-13

INSERT A

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement), and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

INSERT B

"In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the secondary containment or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of secondary containment when fuel being handled is recently irradiated, i.e., fuel that has occupied part of the critical reactor core within the previous 24 hours.

During handling of fuel and CORE ALTERATIONS, secondary containment and FRVS actuation is not required. However, building ventilation will be operating during fuel handling and CORE ALTERATIONS and will be capable of drawing air into the building and exhausting through a monitored pathway. To reduce doses even further below that provided by 24 hours of natural decay, a single normal or contingency method to promptly close secondary containment penetrations is provided in accordance with RG 1.183. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of the "prompt methods" (defined as within 30 minutes) is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored. These contingencies are to be utilized after a postulated fuel handling accident has occurred to reduce doses even further below that provided by the natural decay.

CORE ALTERATION

Insert A →

~~1.7 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement) are not considered to be CORE ALTERATIONS. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be highest value of the FLPD which exists in the core.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC-approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

TABLE 3.3.2-1 (Continued)

NOTES

*	When handling recently irradiated fuel in the secondary containment and during ~~CORE ALTERATIONS and~~ operations with a potential for draining the reactor vessel.
**	When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.
#	Refer to Specification 3.1.5 for applicability.
##	The hydrogen water chemistry (HWC) system shall not be placed in service until reactor power reaches 20% of RATED THERMAL POWER. After reaching 20% of RATED THERMAL POWER, and prior to operating the HWC system, the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER and after the HWC system has been shutoff, the background level and associated setpoint shall be returned to the normal full power values. If a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.
(a)	A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b)	Also trips and isolates the mechanical vacuum pumps.
(c)	Also starts the Filtration, Recirculation and Ventilation System (FRVS).
(d)	Refer to Table 3.3.2-1 table notation for the listing of which valves in an actuation group are closed by a particular isolation signal. Refer to Tables 3.6.3-1 and 3.6.5.2-1 for the listings of all valves within an actuation group.
(e)	Sensors arranged per valve group, not per trip system.
(f)	Closes only RWCU system isolation valve(s) HV-F001 and HV-F004.
(g)	Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.
(h)	Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.
(i)	Manual isolation closes HV-F003 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.
(j)	Trip functions common to RPS instrumentation.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
h. HPCI Torus Compartment Temperature - High	NA	Q	R	1, 2, 3
i. Drywell Pressure - High	NA	Q	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	Q	R	1, 2, 3
c. Manual Initiation	NA	Q(a)	NA	1, 2, 3

* When handling recently irradiated fuel in the secondary containment and during ~~CORE ALTERATIONS and~~ operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

\# Refer to Specification 3.1.5 for applicability.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 92 days.

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the reactor building is at a negative pressure.

b. Verifying at least once per 31 days that:

    1. All secondary containment equipment hatches and blowout panels are closed and sealed.

    2. a. For double door arrangements, at least one door in each access to the secondary containment is closed.

        b. For single door arrangements, the door in each access to the secondary containment is closed except for routine entry and exit.

    3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

    1. Verifying that four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the filtration recirculation and ventilation system will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 375 seconds, and

*When recently irradiated fuel is being handled in the secondary containment and during ~~CORE ALTERATIONS and~~ operations with a potential for draining the reactor vessel.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment ventilation system (RBVS) automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable dampers to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated damper secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifying the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation damper actuates to its isolation position.

c. By verifying the isolation time to be within its limit at least once per 92 days.

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* When recently irradiated fuel is being handled in the secondary containment and during ~~CORE ALTERATIONS and~~ operations with a potential for draining the reactor vessel.

CONTAINMENT SYSTEMS

3.6.5.3 FILTRATION, RECIRCULATION AND VENTILATION SYSTEM (FRVS)
FRVS VENTILATION SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3.1 Two FRVS ventilation units shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

a. With one of the above required FRVS ventilation units inoperable, restore the inoperable unit to OPERABLE status within 7 days, or:

1. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In Operational Condition *, place the OPERABLE FRVS ventilation unit in operation, or suspend handling of recently irradiated fuel in the secondary containment ~~, CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both ventilation units inoperable in Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment ~~or CORE ALTERATIONS~~ or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Each of the two ventilation units shall be demonstrated OPERABLE:

a. At least once per 14 days by verifying that the water seal bucket traps have a water seal and making up any evaporative losses by filling the traps to the overflow.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 15 minutes ~~10 hours with the heaters on in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters~~.

*When recently irradiated fuel is being handled in the secondary containment and during ~~CORE ALTERATIONS and~~ operations with a potential for draining the reactor vessel.

~~**The provisions of Specification 3.0.4 are not applicable for initiation of handling of irradiated fuel in the secondary containment and CORE ALTERATIONS provided the plant in OPERATIONAL CONDITION 5, with reactor water level equal to or greater than 22 feet 2 inches.~~

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or upon determination** that the HEPA filters or charcoal adsorbent could have been damaged by structural maintenance or adversely affected by any chemicals, fumes or foreign materials (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 9,000 cfm ± 10% for each FRVS ventilation unit.

   2. Verifying within 31 days after removal from the FRVS ventilation units, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than ~~2.~~5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity ~~70%~~. 95%

   3. Verifying a subsystem flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FRVS ventilation units, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than ~~2.~~5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of ~~70%~~. 95%

**This determination shall consider the maintenance performed and/or the type, quantity, length of contact time, known effects and previous accumulation history for all contaminants which could reduce the system performance to less than that verified by the acceptance criteria in items c.1 through c.3 below.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5 inches Water Gauge in the ventilation unit while operating the filter train at a flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit.

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Manual initiation from the control room, and

   b. Simulated automatic initiation signal.

3. ~~Verifying that the heaters dissipate 32 ± 3 kw for each ventilation unit when tested in accordance with ANSI N510-1980, and verifying humidity is maintained less than or equal to 70% relative humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation.~~

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2 March 1978, while operating the system at a flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit.

CONTAINMENT SYSTEMS

3.6.5.3 FILTRATION, RECIRCULATION AND VENTILATION SYSTEM (FRVS)
FRVS RECIRCULATION SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3.2 Six FRVS recirculation units shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

a. With one or two of the above required FRVS recirculation units inoperable, restore all the inoperable unit(s) to OPERABLE status within 7 days, or:

1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. ~~**~~ The provisions of Specification 3.0.3 are not applicable.

b. With three or more of the above required FRVS recirculation units inoperable in Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment, ~~CORE ALTERATIONS~~ or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

c. With three or more of the above required FRVS recirculation units inoperable in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.2 Each of the six FRVS recirculation units shall be demonstrated OPERABLE:

a. At least once per 14 days by verifying that the water seal bucket traps have a water seal and making up any evaporative losses by filling the traps to the overflow.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters ~~and charcoal adsorbers~~ and verifying that the subsystem operates for ~~at least~~ 15 minutes ~~10 hours with the heaters OPERABLE in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters~~.

*When recently irradiated fuel is being handled in the secondary containment and during ~~CORE ALTERATIONS and~~ operations with a potential for draining the reactor vessel.

~~**The provisions of Specification 3.0.4 are not applicable for initiation of handling of irradiated fuel in the secondary containment and CORE ALTERATIONS provided the plant is in OPERATIONAL CONDITION 5, with reactor water level equal to or greater than 22 feet 2 inches.~~

c. At least once per 18 months or upon determination** that the HEPA filters ~~or charcoal adsorbent~~ could have been damaged by structural maintenance or adversely affected by any ~~chemicals, fumes or~~ foreign materials ~~(1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem~~ by:

1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a and C.5.c ~~and C.5.d~~ of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 30,000 cfm ± 10% for each FRVS recirculation unit.

~~2. Verifying within 31 days after removal from the FRVS recirculation units, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyliodide penetration less than 10.0% when tested in accordance with ATM D3803-1989 at a temperature of 30°C and a relative humidity 70%.~~

2.~~3.~~ Verifying a subsystem flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit during system operation when tested in accordance with ANSI N510-1980.

d. ~~After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FRVS recirculation units, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 10.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 70%.~~ not used

**This determination shall consider the maintenance performed and/or the type, quantity, length of contact time, known effects and previous accumulation history for all contaminants which could reduce the system performance to less than that verified by the acceptance criteria in items c.1 ~~through~~ and c.~~3~~2 below.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the ~~combined HEPA filters and charcoal adsorber banks~~ exhaust duct, is less than 8 inches Water Gauge in the recirculation filter train while operating the filter train at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Manual initiation from the control room, and

   b. Simulated automatic initiation signal.

3. ~~Verifying that the heaters dissipate 100 ± 10 kw for each recirculation unit when tested in accordance with ANSI N510-1980, and verifying humidity is maintained less than or equal to 70% relative humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation.~~

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2 March 1978, while operating the system at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.

g. ~~After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.~~

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

SAFETY AUXILIARIES COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent safety auxiliaries cooling system (SACS) subsystems, with each subsystem comprised of:

a. Two OPERABLE SACS pumps, and

b. An OPERABLE flow path consisting of a closed loop through the SACS heat exchangers and SACS pumps and to associated safety related equipment shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5, and ** the subsystems associated with systems and components required OPERABLE by Specification 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1 and 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and **.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. a. With one SACS pump inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.*** If the condition specified by *** can not be met, be in at least HOT SHUTDOWN within the next 72 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one SACS heat exchanger inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore the heat exchanger to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN with the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one SACS subsystem otherwise inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, realign at least one of the affected diesel generators to the OPERABLE SACS subsystem within 2 hours, within 6 hours realign other affected SACS supported loads required to support plant operation for at least 72 hours, and restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump and heat exchanger within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.***

** When handling recently irradiated fuel in the secondary containment.

*** Two diesel generators and two service water pumps associated with the unaffected SACS loop must be OPERABLE.

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent station service water system loops, with each loop comprised of:

a. Two OPERABLE station service water pumps, and

b. An OPERABLE flow path capable of taking suction from the Delaware River (ultimate heat sink) and transferring the water to the SACS heat exchangers,

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two loops.

b. In OPERATIONAL CONDITION 4, 5 and *, one loop.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one station service water pump inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.** If the condition specified by ** can not be met, be in at least HOT SHUTDOWN within the next 72 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one station service water pump in each loop inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore at least one inoperable pump to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.***

3. With one station service water system loop otherwise inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, assess the operability of the associated SACS loop and take the ACTION specified in LCO 3.7.1.1, Action Statement a.2, if required, and restore the inoperable station service water system loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.**

* When handling recently irradiated fuel in the secondary containment.

** Two diesel generators and two SACS pumps associated with the unaffected service water loop must be OPERABLE.

*** Two diesel generators and SACS pumps associated with the required OPERABLE service water pumps and all SACS heat exchangers must be OPERABLE.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The ultimate heat sink (Delaware River) shall be OPERABLE with:

a. A minimum river water level at or above elevation -9'0 Mean Sea Level, USGS datum (80'0 PSE&G datum), and

b. An average river water temperature of less than or equal to 85.0°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

With the river water temperature in excess of 85.0°F, continued plant operation is permitted provided that both emergency discharge valves are open and emergency discharge pathways are available. With the river water temperature in excess of 88.0°F, continued plant operation is permitted provided that all of the following additional conditions are satisfied: ultimate heat sink temperature is at or below 89.0°F, all SSWS pumps are OPERABLE, all SACS pumps are OPERABLE, all EDGs are OPERABLE and the SACS loops have no cross-connected loads (unless they are automatically isolated during a LOP and/or LOCA); otherwise, with the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITIONS 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITIONS 4 or 5, declare the SACS system and the station service water system inoperable and take the ACTION required by Specification 3.7.1.1 and 3.7.1.2.

c. In Operational Condition *, declare the plant service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The ultimate heat sink shall be determined OPERABLE:

a. By verifying the river water level to be greater than or equal to the minimum limit at least once per 24 hours.

b. By verifying river water temperature to be within its limit:

1) at least once per 24 hours when the river water temperature is less than or equal to 82°F.

2) at least once per 2 hours when the river water temperature is greater than 82°F.

* When handling recently irradiated fuel in the secondary containment.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency filtration system subsystems shall be OPERABLE with each subsystem consisting of:

a) One control room supply unit,
b) One filter train, and
c) One control room return air fan.

APPLICABILITY: All OPERATIONAL CONDITIONS 1,2,3,4 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

1. With one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the pressurization/recirculation mode of operation.

2. With both control room emergency filtration subsystems inoperable, suspend CORE ALTERATIONS, handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency filtration subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F#.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, the control area chilled water pump, flow

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*When recently irradiated fuel is being handled in the secondary containment.

#This does not require starting the non-running control emergency filtration subsystem.

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators, one of which shall be diesel generator A or diesel generator B, each with:

1. A separate fuel oil day tank containing a minimum of 360 gallons of fuel.

2. A fuel storage system consisting of two storage tanks containing a minimum of 44,800 gallons of fuel.

3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22'-2" above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

c. With one fuel oil transfer pump inoperable, realign the flowpath of the affected tank to the tank with the remaining operable fuel oil transfer pump within 48 hours and restore the inoperable transfer pump to OPERABLE status within 14 days, otherwise declare the affected emergency diesel generator (EDG) inoperable. This variance may be applied to only one EDG at a time.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

* When handling recently irradiated fuel in the secondary containment.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four channels of the D.C. electrical power sources, one of which shall be channel A or channel B, shall be OPERABLE with:

a. Channel A, consisting of:
   1. 125 volt battery 1AD411
   2. 125 volt full capacity charger# 1AD413 or 1AD414

b. Channel B, consisting of:
   1. 125 volt battery 1BD411
   2. 125 volt full capacity charger# 1BD413 or 1BD414.

c. Channel C, consisting of:
   1. 125 volt battery 1CD411
   2. 125 volt full capacity charger# 1CD413 or 1CD414
   3. 125 volt battery 1CD447
   4. 125 volt full capacity charger 1CD444

d. Channel D, consisting of:
   1. 125 volt battery 1DD411
   2. 125 volt full capacity charger# 1DD413 or 1DD414
   3. 125 volt battery 1DD447
   4. 125 volt full capacity charger 1DD444

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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*When handling recently irradiated fuel in the secondary containment.
#Only one full capacity charger per battery is required for the channel to be OPERABLE.

4

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than two channels of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system channels shall be determined energized at least once per 7 days by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

*When handling recently irradiated fuel in the secondary containment.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the refueling floor due to a fuel handling accident. When Exhaust Radiation - High is detected, secondary containment isolation and actuation of the FRVS are initiated to limit the release of fission products as assumed in the UFSAR safety analyses (Ref. 4).

The Exhaust Radiation - High signals are initiated from radiation detectors that are located on the ventilation exhaust ducts coming from the reactor building and the refueling floor zones, respectively. Three channels of Reactor Building Exhaust Radiation - High Function and three channels of Refueling Floor Exhaust Radiation - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Refueling Floor and Reactor Building Exhaust Radiation - High Functions are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In OPERATIONAL CONDITIONS 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these OPERATIONAL CONDITIONS; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during OPDRVs, when handling recently irradiated fuel in the secondary containment ~~and during CORE ALTERATIONS,~~ because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

The valve groups actuated by this Function are listed in Table 3.3.2-1.

2.e. Manual Initiation

The Manual Initiation for secondary containment isolation can be performed by manually initiating a primary containment isolation. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

There are four push buttons for the logic, one manual initiation push button per PCIS channel. There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of Manual Initiation Function are available and are required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2, and 3, and during OPDRVs, when handling recently irradiated fuel in the secondary containment. ~~and during CORE ALTERATIONS.~~ These are the OPERATIONAL CONDITIONS and other specified conditions in which the Secondary Containment Isolation automatic Functions are required to be OPERABLE.

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR is more conservative than the Inservice Testing Program requirements.

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of 0.25 psid is valid. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this unit, the 18 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a 0.25 inch water gage vacuum in the reactor building with the filtration recirculation and ventilation system (FRVS) once per 18 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

Insert B

~~The OPERABILITY of the FRVS ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses and with the drawdown analysis. Continuous operation of the system with the heaters and humidity control instruments OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.~~