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MAR 2 8 2003

LR-N03-0086

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

CORRECTION OF INFORMATION PROVIDED IN REQUEST FOR LICENSE AMENDMENT SALEM GENERATING STATION, UNIT NOS. 1 AND 2 FACILITY OPERATING LICENSE DPR-70 AND DPR-75 DOCKET NOS. 50-272 AND 50-311

On June 28, 2002, PSEG Nuclear submitted a request for license amendment via letter LR-N02-0231. This request was approved by the NRC in Amendments 251 and 232 to the Salem Unit 1 and 2 Technical Specifications, respectively.

In the amendment request, PSEG provided tables of dose information concerning the results of fuel handling accident (FHA) occurring in the Containment Building and a FHA occurring in the Fuel Handling Building. In each of these tables, the exclusion area boundary (EAB) dose result reported was 4.15E+00 rem. An error was subsequently identified in the calculation for the dose analysis. This error identifies that the EAB dose reported in letter LR-N02-0231 was off by a factor of 10 in the conservative direction. The actual EAB dose should have been stated as 4.15E-01 rem. The FHA dose calculation has been revised to correct this error. Based on the conservative nature of the dose calculation error, PSEG has reviewed the NRC's safety evaluation report associated with Amendments 251 and 232 and determined that there is no impact to the conclusions made by the NRC. The new dose calculated provides additional margin to the dose limits provided in 10CFR50.67 and Regulatory Guide 1.183.

The error identified in the dose calculation is associated with the X/Q value for the EAB. The actual X/Q value for the EAB is 1.30E-4. Based on this change in the X/Q value, the information contained on page 2 of the Attachment to the NRC's SER for Amendments 251 and 232 needs to be revised to reflect the corrected X/Q value for the EAB.

A001

If you have any questions concerning this transmittal, please contact Brian Thomas of my staff at 856-339-2022.

Sincerely,

Gabor Salamon
Nuclear Safety and Licensing Manager

C Mr. H. J. Miller, Regional Administrator
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File

JUN 2 8 2002

LR-N02-0231
LCR S02-03

PSEG
Nuclear LLC

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Gentlemen:

REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
REFUELING OPERATIONS – FUEL DECAY TIME PRIOR TO COMMENCING CORE ALTERATIONS OR MOVEMENT OF IRRADIATED FUEL
SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications (TS) for the Salem Nuclear Generating Station, Units 1 and 2. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG proposes to revise the requirements for Fuel Decay Time prior to commencing movement of irradiated fuel. TS 3/4.9.3 "Decay Time" is revised to allow fuel movement in the containment to commence 100 hours after reactor subcriticality between October 15th through May 15th. Should refueling occur between May 16th and October 14th, the current 168 hours decay time limit will remain in place. This amendment, when approved, will remain valid through the year 2010. PSEG will then reanalyze the SFP loading conditions and determine the required licensing actions beyond 2010. A similar amendment was approved and issued for American Electric Power's, DC Cook Units 1 & 2 in November 2001.

The bases for these changes are:

1. PSEG has performed re-analysis of the atmospheric dispersion factors and radiological doses to the members of the public and control room personnel by applying the guidelines contained in 10CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term. Therefore, the proposed change also requests NRC approval of selective implementation of Alternative Source Term methodology for the Salem Units 1 & 2 Fuel Handling Accident Inside Containment and within the Spent Fuel Handling Building. The Salem Units 1 & 2 UFSAR will be updated to reflect the amendment and analysis following NRC approval.
2. PSEG has performed Spent Fuel Pool (SFP) re-analysis to determine the capability of the cooling system to maintain Spent Fuel Pool water temperature below the analyzed limit of 180°F, which prevents degradation of the pool liner, after 100 hours decay time prior to fuel transfer.
3. PSEG will continue the application of the Spent Fuel Pool Integrated Decay Heat Management Program developed and described in PSEG letter (LR-N96218) to the NRC dated August 2, 1996. This program is designed to perform heat load calculations prior to core fuel offloads and estimate the required cooling requirements to ensure that the Spent Fuel Pool water temperature limit of 180°F is not exceeded.

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. This amendment to the Salem TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

An evaluation of the requested changes is provided in Attachment 1 to this letter.

The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 2.

PSEG requests approval of the proposed License Amendment by September 12, 2002 to be implemented within 30 days. This will allow the start of Salem Nuclear Generating Station Unit 1, fifteenth (1R15) refueling outage that is scheduled for October 12, 2002. PSEG will implement the amendment within 30 days of issuance to coincide with the start of the refueling outage.

No new commitments are made in this submittal.

Should you have any questions regarding this request, please contact Mr. Brian Thomas at 856-339-2022.

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

Sincerely,

Executed on June 28, 2002

D. F. Garchow
Vice President - Operations

Attachments (2)

C: Mr. H. J. Miller, Administrator - Region I
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CORRESPONDENCE CONTROL PROGRAM
STANDARD INTERNAL DISTRIBUTION

DATE: 6/28/02 **DOCUMENT DATE:** 6/28/02

DOCUMENT TYPE: LCR S02-03

Attached: LR-N02-0231

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SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS REFUELING OPERATIONS – FUEL DECAY TIME PRIOR TO MOVEMENT OF IRRADIATED FUEL

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS REFUELING OPERATIONS – FUEL DECAY TIME PRIOR TO COMMENCING CORE ALTERATIONS OR MOVEMENT OF IRRADIATED FUEL

Table of Contents

1. DESCRIPTION 1

2. PROPOSED CHANGE 1

3. BACKGROUND 1

4. TECHNICAL ANALYSIS 1

4.1 Spent Fuel Pool Handling System Capability 2

4.2 Fuel Handling Accident Alternative Source Term 5

Table 1 - Fuel Handling Accident Analysis Assumptions 12

Table 2- Core Inventory 12

Figure 1- Plant Layout

5. REGULATORY SAFETY ANALYSIS 13

5.1 No Significant Hazards Consideration 13

5.2 Applicable Regulatory Requirements/Criteria 15

6. ENVIRONMENTAL CONSIDERATION 16

7. REFERENCES 17

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS REFUELING OPERATIONS – FUEL DECAY TIME PRIOR TO COMMENCING CORE ALTERATIONS OR MOVEMENT OF IRRADIATED FUEL

1. DESCRIPTION

Current Requirements

TS 3.9.3 requires that the reactor has been subcritical for at least 168 hours prior to movement of irradiated fuel from the reactor vessel. The action statement requires suspension of all operations involving movement of irradiated fuel within the reactor pressure vessel with a decay time of less than 168 hours. The associated surveillance, TS Surveillance Requirement 4.9.3 requires verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

The supporting Bases for this Specification ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The decay time is consistent with the assumptions used in the accident analyses.

2. PROPOSED CHANGE

The proposed changes to the Technical Specifications and associated Bases would revise TS 3/4.9.3 to allow a 100-hour decay time between October 15th and May 15th, and a 168-hour decay time between May 16th and October 14th. This amendment will remain valid through 2010. At that point, further evaluations will be required to determine the Spent Fuel Pool heat load and temperature limitations. The respective action statement is revised to replace "168 hours" with the " required decay time". The Surveillance requirements TS 4.9.3 is being revised from "for at least 168 hours" with "as required". The TS Bases 3/4.9.3 is revised to add additional details regarding the new decay time requirements and the application of the Integrated Decay Heat Management Program to limit Spent Fuel Pool temperature below 180°F.

Figure 1 depicts an illustration of structures and elevations provided to assist in the review of this proposed change.

The marked up Technical Specification pages are included in Attachment 2.

3. BACKGROUND

The 168-hour decay time was included in the Salem TS with Amendment 151 and 131 to DPR-70 and DPR-75, respectively, on May 4, 1994. These amendments were associated with the Spent Fuel Pool Reracking providing an additional 10 years of spent fuel storage.

4. TECHNICAL ANALYSIS

In order to evaluate the reduction in the required decay time from 168 hours to 100 hours, two evaluations were performed: Spent Fuel Pool Cooling Capacity and Fuel Handling Accident (FHA) Dose Assessments. Summaries of these two evaluations are provided below:

4.1 Spent Fuel Pool Cooling System Capability with Core Offload Starting 100-hours After Shutdown.

The current 168-hour limit is based upon the capability of the Spent Fuel Pool cooling system when River temperatures, and the consequent Component Cooling Water (CCW) temperatures, are at their highest. These analyses considered the Delaware River water temperature to be at 90°F, with CCW at 99°F. This condition has never occurred at Salem, but if it did, it would more likely occur in late July or early August, when Delaware River temperatures typically peak.

While the 168-hour delay conservatively covers the entire year, it imposes an unnecessary penalty on plant operators in the cooler months, when refuelings are typically scheduled.

In view of the above, this evaluation considers Spent Fuel Pool cooling capabilities if a 100-hour delay is imposed prior to movement of irradiated fuel in the reactor vessel, and is restricted to movement of irradiated fuel in the reactor vessel that occur between October 15th and May 15th.

The significant assumptions for this analysis are:

1. Reactor power is conservatively considered to be 3479 MWt (1.02 x 3411 MWt). This envelopes the current 3459 MWt based upon the 1.4% power uprate.
2. Based on current refueling programs, fuel assemblies while in the reactor vessel will be assumed to be expended in accordance with the following:
   - 76 assemblies with 17 months of effective full power operation
   - 76 assemblies with 34 months of effective full power operation
   - 41 assemblies with 51 months of effective full power operation.
3. Defueling of 193 assemblies will be assumed to require 46 hours as per current scheduling. Therefore defueling is complete 146 hours (6.08 days) after shutdown (100 hours + 46 hours). Actual defueling times for the past 5 Salem outages are listed below:

Refueling Cycle	Defueling Times
1R13	60 hours
1R14	53 hours
2R10	58 hours
2R11	53 hours
2R12	53 hours

4. There are currently 920 fuel assemblies in the Unit 1 Spent Fuel Pool (as of 1R14 in April 2001) and 812 elements in the Unit 2 pool (as of 2R12 in April 2002). Therefore, Unit 1 SFP heat load is considered to bound Unit 2 SFP.
5. The maximum number of fuel elements that can be loaded into a Salem Spent Fuel Pool is 1632, as described in TS 5.6.3.

6. Background heat in the Unit 1 SFP at any given refueling between the present and end of life (or full pool) is assumed to be a straight line between 2.31 x 10^6 Btu/hour (prior to outage 1R13) and 8.46 x 10^6 Btu/hour (end of life).
7. Net thermal capacity of Spent Fuel Pool water at the end of life with all fuel racks filled (thereby minimizing available water volume) is 1.96 x 10^6 Btu/°F.

The basic parameters that are used are reiterated below:

1. Refueling operations are conducted during the period of October 15th to May 15th.
2. All 193 fuel assemblies are off-loaded to the Spent Fuel Pool (full core offload).
3. In addition to the fresh 193 assemblies, the background heat (old assemblies) in the pool represents the background heat that will exist in the year 2010.
4. Delaware River water temperatures are determined from 30 years of historical data.
5. Defueling begins 100 hours after reactor shutdown (subcritical).
6. All Spent Fuel Pool heat removal is via the Spent Fuel Pool Cooling System. No credit is taken for heat transfer via evaporative cooling or to the Spent Fuel Pool (concrete) structure.

The methodology applied in the evaluation is described below:

1. Determine the decay heat rate from the off-loaded core using USNRC Branch Technical Position ASB 9-2.
2. Determine background heat that will exist in the Spent Fuel Pool in the year 2010.
3. Evaluate Delaware River temperatures during the period from October through May.
4. Determine CCW temperature based on river water temperature and SFP heat load.
5. Evaluate the ability of the Spent Fuel Pool Cooling System to maintain pool temperature limits.

It is assumed that heat removal from the Salem Spent Fuel Pools uses only forced cooling provided by the Spent Fuel Pool heat exchangers. By relying exclusively on the Spent Fuel Pool heat exchangers, several resulting conservatisms are described below:

1. No credit is taken for evaporative cooling, i.e. pool bulk temperature cooling resulting from evaporation at the surface of the Spent Fuel Pool. Previous analyses indicate that evaporative cooling contributes 0.86 x 10^6 Btu/hour at 150°F and 3.87 x 10^6 Btu/hour at 180°F. Consequently, if the pool reaches 180°F, evaporative cooling amounts to nearly 10% of the peak heat load in the Spent Fuel Pool.
2. No credit is taken for cooling through the concrete structure of the pool. Heat is conducted through the pool steel liner, concrete structure, and ultimately to the cooler environment beyond the structure. The higher the pool water temperature, the more heat transmitted through the structure.

3. RHR cooling continues to provide forced cooling to the Spent Fuel Pool with all fuel elements removed to the Spent Fuel Pool as long as the refueling canal remains flooded and the transfer gate is open. The cooler water in the reactor vessel and refueling canal will transfer to the Spent Fuel Pool via natural circulation through the transfer gate. This potential cooling source is not credited herein.

The results of the Spent Fuel Pool Cooling System Capability Evaluation are provided below:

The residual heat from the 193-assembly offload to the Spent Fuel Pool is shown to be 3.72 x 10^7 Btu/hr as summarized in the following table. The 146 hours after shutdown includes the 100-hour delay plus an additional 46 hours to offload the 193 assemblies. A 10% uncertainty factor is included.

This is the highest heat load in the pool from the newly discharged core, and it exists only at the moment that the final assembly is moved into the pool. After that time, the heat load continuously decays to lower values. Nonetheless, this value is used throughout this evaluation as the heat load in the Spent Fuel Pool.

Number of Assemblies	Reactor Power	Time to Off-Load After Shutdown	Effective Full Power Hours of Burnup	Calculated Decay Heat
76	3479 MWt	6.08 days (146 hrs)	12,410 (17 mos)	1.31 x 10^7 Btu/hr
76	3479 MWt	6.08 days (146 hrs.)	24,820 (34 mos.)	1.36 x 10^7 Btu/hr
41	3479 MWt	6.08 days (146 hrs)	37,230 (51 mos.)	7.43 x 10^6 Btu/hr
Heavy Elements (all assemblies)	3479 MWt	6.08 days (146 hrs)	Same as above	3.03 x 10^6 Btu/hr
Core Total				**3.72 x 10^7 Btu/hr**
Background heat				6.8 x 10^6 Btu/hr
Peak Pool Heat Load in 2010				**4.4 x 10^7 Btu/hr**

The average monthly temperature in the Delaware River (measured at Reedy Island) between the months of October and May are 63°F or lower. To this is added 3°F, the historical difference between Reedy Island and the Salem intake, resulting in a 66°F inlet temperature. These temperatures are based upon 30 years of weekly data recorded at Reedy Island, a location just upstream of Salem and Hope Creek.

The CCW supply temperature of 71°F is based on a Service Water inlet temperature of 66°F.

In the final analysis, both pools can be maintained at or below the design SFP temperature limit of 180°F. Actual operational requirements will be determined on an outage-by-outage basis by the performance of an assessment of Spent Fuel Pool heat loads in accordance with the Integrated Decay Heat Management Program.

Conclusion:

During the period from October 15th through May 15th up to and including the year 2010, a fully radiated 193 element core can be off-loaded to a Spent Fuel Pool with a 100-hour in-vessel decay, rather than a 168 hour decay, because the Spent Fuel Pool system is capable of maintaining both Salem pools below the design SFP temperature of 180°F as described in UFSAR Section 9.1.3.1.

This conclusion is justified because: (1) the Salem Outage Risk Management Program which includes Spent Fuel Pool Integrated Decay Heat Management Program, requires a pre-outage assessment of the Spent Fuel Pool heat loads and heatup rates to assure available Spent Fuel Pool capability prior to offloading fuel and, (2) the inherent conservatisms in this calculation provide for additional cooling sources that are not credited herein.

4.2 Fuel Handling Accident - Alternative Source Term Analysis

The purpose of this analysis is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and Control Room (CR) doses due to a fuel handling accident (FHA) occurring in the containment building and in the Fuel Handling Building (FHB). The FHA analyses are performed using a selective implementation of the Alternative Source Term (AST), guidance in the Regulatory Guide 1.183, Appendix B, and TEDE dose criteria. Additional conservatism was used by assuming no containment closure during fuel movement. These additional conservative assumptions will be used for a future amendment presently being developed by PSEG to relax the containment closure requirements during fuel movement.

Fuel handling accidents are postulated in the containment and FHB with the reactor being subcritical for at least 96 hours. Conservative assumptions are used in that; activity is released to the environment through the opened Containment Equipment Hatch (CEH) or the plant vent (PV).

The FHA is analyzed using the plant specific design inputs supporting the proposed licensing bases. The design inputs are compatible to the AST and TEDE dose criteria.

The scrubbing of the activity in the reactor cavity and spent fuel storage pool are credited in the analyses. The scrubbing effects are provided by the 23 feet height of water over the top of the Reactor Pressure Vessel (RPV) flange and over the top of the irradiated fuel assemblies in the Spent Fuel Pool storage racks.

The core inventory is calculated based on thermal power level of 3,600 MW_t. The thermal power level of 3,600 MW_t is used to provide a margin over rated thermal power level of 3,459 MW_t for future power uprate. The core activity is shown in Table 2. The activity released during the FHA is based on a drop of one fuel assembly with a radial peaking of 1.70 (conservatively used instead of 1.65) and all fuel rods in the assembly sustaining cladding rupture.

ASSUMPTIONS

The regulatory requirements in the Regulatory Guide 1.183, Appendix B are adopted as assumptions, which are incorporated as design inputs along with other plant-specific as-built design parameters.

Table 1 contains a summary of assumptions for ease of reference.

Credit for Engineered Safeguard Features

Credit is taken only for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The normal Control (CR) air intake monitors are required to be operable by TS 3.3.3.1 in ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS. The normal CR air intake monitor's function of preferential alignment of the less contaminated outside air emergency intake is conservatively not credited. The Control Room Emergency Air Conditioning System (CREACS) charcoal filtration operation is credited with a 2-minutes system response delay. The FHB safety related charcoal filtration system is conservatively not credited in the analysis.

Source Term Assumptions

- Per Regulatory Guide 1.183, Regulatory Position 3.2, for non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3 of RG 1.183. The release fractions are incorporated in Table 3 in conjunction with the core fission product inventory with the maximum core radial peaking factor of 1.70 and the core inventory at 3,600 MWt power level. The bromines are neglected from thyroid dose consideration due to their low thyroid dose conversion factors; relatively short half-lives, and decay into insignificant daughters.

- Per Regulatory Guide 1.183, Appendix B, Regulatory Position B.1.1, the number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. The Fuel Handling Accident analysis described in the UFSAR Section 15.4.6 assumes to result in the release of the gaseous fission products contained in the fuel cladding gaps of all the fuel rods in a peak-power fuel assembly (radial peaking factor of 1.70). There have been several industry events where fuel bundles have been dropped during fuel handling. In each case, the actual releases from fuel have been minimal or nonexistence. This evidence shows that the assumption of damage of one fuel assembly in the radiological analysis for a FHA is conservative.

- Per Regulatory Guide 1.183, Appendix B, Regulatory Position B.1.2, the fission product release from the breached fuel is based on fraction of fission product inventory in gap and the estimate of the number of fuel rods breached.

Core Inventory

The inventory of fission products in the reactor core and available for gap release from damaged fuel is based on the maximum power level of 3,600 MWt corresponding to current fuel enrichment and fuel burnup. All the gap activity in the damaged rods is assumed to be instantaneously released. The radionuclides included are xenons, kryptons, and iodines. The fraction of fission product in gap activity is shown below:

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

It is further assumed that irradiated fuel shall not be removed from the reactor until the unit has been sub-critical for at least 96 hours. (Proposed amendment for Decay Time is 100 hours)

Timing of Release Phase

The release from the fuel gap and the fuel pellet is assumed to occur instantaneously with the onset of the projected damage.

Chemical Form

The chemical form of radioiodine released from the fuel to the surrounding water should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously.

Water Depth

The decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species

Noble Gases

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

FHA Occurring In Containment Building

The CEH provides direct release path to the environment. The personnel air locks and piping penetrations provide release paths to the environment through the plant vent via piping penetration area.

The comparison of the χ/Qs for both units, indicate that the post-FHA release from Unit 1 CEH produces the highest χ/Qs. Therefore, the only two sets of the most limiting atmospheric dispersion factors for these release paths are compared in the following table.

Time Interval (hr)	Salem 1 CR Intake χ/Qs (s/m3)	
	Unit 1 Equipment Hatch Unit 1 CR Intake	Unit 1 Plant Vent Unit 1 CR Intake
0-2	2.86E-03	1.78E-03
2-8	2.22E-03	1.31E-03
8-24	9.15E-04	5.22E-04
24-96	6.60E-04	3.77E-04
96-720	5.62E-04	3.17E-04

The comparison of χ/Qs in the above table indicates that the CEH provides a conservative release path for the FHA occurring in the containment. Therefore, the EAB, LPZ, and CR doses are calculated using the post-FHA release through the CEH. Activity release rate from the CEH is calculated based on the removal of 99% of radioactive material released from the damaged fuel to the environment over a 2-hour period. The resulting doses at the EAB, LPZ, and CR locations are compared with the regulatory allowable limits in page 10.

FHA Occurring In Fuel Handling Building

A parametric study is performed to determine a conservative release model using a post-FHA release rate based on 0-2 hour release and a rapid release rate of one FHB volume per minute. The results of the parametric study indicate that the post-FHA release over two hours period yields a higher CR dose, due to a larger amount of activity entering the CR volume. Should a FHA occur in the FHB, activity can be released through either the plant vent or FHB rollup door at ground level. However, the following post-FHA release paths are identified during the FHB pressurization due to a failure of one (1) FHB exhaust fan:

1. Release through the plant vent at a rate of 15,300 cfm
2. Release through truck bay at a rate of 3,883 cfm
3. Release through gravity damper 256 cfm

The atmospheric dispersion factors (χ/Qs) for the plant vent and FHB roll-up door are calculated using the ARCON96 computer code. The χ/Qs for gravity damper release are conservatively assumed to be the same as those for a smoke hatch. The smoke hatch χ/Qs were developed using ARCON95 computer code. Since the FHA release duration is two hours and the values of 0-2 χ/Q is not impacted by ARCON96 analysis, the smoke hatch 0-2 hr χ/Q value is used to calculate the equivalent 0 to 2 hr χ/Q for a combined post-FHA release path. The equivalent χ/Q is used with the post-FHA unfiltered release from the FHB to calculate the EAB, LPZ, and CR doses.

Activity from the FHB is assumed to be released to the environment at a rate of 21,439 cfm (design flow rate of 2 exhaust fans + 10%). The resulting doses at the EAB, LPZ, and CR locations are compared with the regulatory allowable limits in page 11.

Post-FHA Technical Support Center (TSC) Habitability

The TSC habitability is evaluated to determine the post-FHA TSC dose. The TSC is located in the Clean Facilities Building (CFB) at second and third floors. The CFB is located southeast of the Unit 1 containment building (See Figure 1). As discussed previously, the CEH and PV are the release points for the FHA occurring in the containment and FHB respectively. The TSC emergency air intake is in the Mechanical Equipment Room located on the roof of CFB. The TSC is located closer to Unit 1 containment compared to the Unit 2 containment. The CR doses are considered bounding for TSC for the FHA occurring in the containment and FHB because:

1. The TSC intake is located farther from the subject release points in comparison to the CR intakes. Therefore, the values of corresponding TSC intake χ/Qs will be lower than CR intake χ/Qs and the resulting post-FHA TSC doses will be lower in the same proportion of χ/Qs values.
2. The comparison of CR χ/Qs indicates that the variation of χ/Qs due to change in wind direction is insignificant. Therefore, the TSC χ/Qs will not be impacted by the different in wind direction for 0-2 hr period.
3. Manning the TSC occurs some time after initiation of the postulated accident. Therefore, at first there will be a period with no occupancy during initial phase of the accident.

CR Intake Monitor Response

There are two radiation monitors in each normal CR air intake duct having an alarm/trip set point listed in TS Table 3.3-6, Item 3a. These monitors are: classified as safety related, required to be operable in all modes and during movement of irradiated fuel assemblies and during CORE ALTERATIONS, powered by emergency power sources and, are instantaneously actuated at predetermined setpoints. The post-FHA activity at the CR air intake will instantaneously reach the Alert/Trip setpoint and actuate the monitors. Therefore, these monitors are credited for automatic initiation of the CR Emergency Air Conditioning System (CREACS). The CR intake monitor preferential alignment of less contaminated air intake is conservatively not credited. The delay associated with the CR intake damper closure time, diesel generator start time, if the loss of offsite power is assumed to occur at the time of damper closure, (Loss of Offsite Power not assumed in the Fuel Handling Accident Analysis but used to make this analysis more conservative), and over-all monitor response time results in a total delay time of less than 1.0 minute. A delay of 2 minutes is conservatively assumed in the analysis for the initiation of the Control Room Emergency Air Conditioning System (CREACS) and the control room envelope isolation.

CONCLUSIONS:

FHA Occurring In Containment

The results of analysis in page 10 indicate that the EAB, LPZ, and CR doses are within allowable limits for a FHA occurring in the containment building without containment closure. The results demonstrate that irradiated fuel can be moved in the reactor pressure vessel after reactor being sub-critical for at least 96 hours (proposed TS limit is 100 hours).

FHA Occurring In Fuel Handling Building

The results of analysis in page 11 indicate that the EAB, LPZ, and CR doses are within allowable limits for a FHA occurring in the fuel handling building without crediting the charcoal filtration in the fuel handling ventilation system.

In summary, PSEG evaluated the impacted analyses, and the results demonstrate that the proposed decay times meet required limits and acceptance criteria. The adequacy of this amendment request is based on:

- Spent Fuel Pool Cooling Capacity and Decay Time limits based on Delaware River Water temperatures using conservative assumptions. The 180°F limit is maintained by using the Integrated Spent Fuel Pool Integrated Decay Heat Management Program as defense in-depth.
- Resulting FHA doses calculated in accordance with the guidelines of Regulatory Guide 1.183 and 10CFR 50.67 are within the limits provided.

RESULTS SUMMARY

1. The results of AST analysis for a FHA occurring in the containment building with the CEH, personnel air locks, and containment penetrations open are summarized in the following table:

	Fuel Handling Accident Occurring In Containment Building TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Calculated Dose With CR Pressurized	2.93E+00	4.15E+00 (0.0 hr)	5.94E-02
Allowable TEDE Limit	5.00E+00	6.30E+00	6.30E+00
	RADTRAD Computer Run No.		
	S96FHA4000	S96FHA4000	S96FHA4000

Significant assumptions used in this analysis:

- CEH, personnel air locks, and containment penetration remain open for the duration of the accident
- Containment closure is not credited in the analysis
- Activity is released to the environment at a rate of 99,800 cfm
- CR envelope is pressurized with actuation of the CREACS following a FHA
- CR monitors' preferential alignment to less contaminated CR intake is not credited
- Worst χ/Qs are used for entire duration of the accident
- CR unfiltered in-leakage of 4,000 cfm is assumed
- Release of all gaseous fission product activity in the gaps of all fuel rods in thehighest-power single fuel assembly
- Reactor Cavity overall effective DF = 200
- Core thermal power = 3,600 MW_t
- Radial Peaking Factor = 1.70

2. The results of AST analysis for a FHA occurring in the fuel handling building with a failure of one (1) FHB exhaust fan are summarized in the following table:

	Fuel Handling Accident Occurring In Fuel Handling Building TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Calculated Dose With CR Pressurized	1.90E+00	4.15E+00 (0.0 hr)	5.93E-02
Allowable TEDE Limit	5.00E+00	6.30E+00	6.30E+00
	RADTRAD Computer Run No.		
	FB96FHA4000	FB96FHA4000	FB96FHA4000

Significant assumptions used in this analysis:

- FHB charcoal filtration is not credited
- Activity is released to the environment at a rate of 21,439 cfm
- CR envelope is pressurized with actuation of the CREACS following a FHA
- CR monitors' preferential alignment to less contaminated CR intake is not credited
- Worst χ/Qs are used for entire duration of the accident
- CR unfiltered in-leakage of 4,000 cfm is assumed
- Release of all gaseous fission product activity in the gaps of all fuel rods in thehighest-power single fuel assembly
- Spent Fuel Pool overall effective DF = 200
- Core thermal power = 3,600 MW_t
- Radial Peaking Factor = 1.70

Table 1
FUEL HANDLING ACCIDENT ANALYSIS ASSUMPTIONS

Parameter	Value Assigned
Reactor Thermal Power, MWt	3459 (actual), 3600 (assumed)
Core Inventory, Ci	(Table 2 below)
Containment Closure	Not assumed
Number of Fuel Assemblies in Core	193
Radial Peaking Factor	1.70
Fuel Rod Gap Fraction	
I-131	0.08
Kr-85	0.10
Other Halogens	0.05
Other Noble Gases	0.05
Alkali Metals	0.12
Iodine Species	
Elemental	99.85%
Organic	0.15%
Particulate	none
Water Depth, ft	23
Overall Effective Decontamination Factor (DF)	200
Charcoal Filter Efficiency (NOTE 1)	
Offsite; FHA in SPENT FUEL POOL	No filtration assumed
Offsite; FHA in CTMT	No filtration assumed
Control Room, either	95%
Building Holdup and Dilution	negligible
Release Duration	Assumed to be released in 2 hours
Control Room Volume, ft^3	81,420
Control Room Flow Rates, cfm	
Normal	1320
Emergency Makeup	2200
Recirculation Flow Rate	5000 assumed
Unfiltered In-Leakage	4000 assumed
Switchover to CR Emergency Ventilation	Automatic ESF function in 2 minutes (No credit taken for the preferential alignment of the outside air emergency intake dampers)
Control Room 0-2 hour x/Q, sec/m^3	
FHA – Unit 1 CEH	2.86E-03
FHA – Unit 1 PV	1.85E-03

NOTE 1- Charcoal filter maintains a SAFETY FACTOR of 2 from the analysis to the tested value of iodine removal.

Table 2
Core Inventory (Ci)

Isotope	Activity	Isotope	Activity	Isotope	Activity
KR-83M	1.20E+07	I-132	1.40E+08	XE-133	2.00E+08
KR-85M	2.60E+07	I-133	2.00E+08	XE-135	5.00E+07
KR-85	1.10E+06	I-134	2.20E+08	XE-135M	4.00E+07
KR-87	4.70E+07	I-135	1.90E+08	XE-138	1.60E+08
KR-88	6.70E+07	XE-131M	7.00E+05		
I-131	9.90E+07	XE-133M	2.90E+07		

5.0 Regulatory Safety Analysis

5.1 Basis for proposed no significant hazards consideration determination

As required by 10 CFR 50.91(a), PSEG provides its analysis of the no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
2. create the possibility of a new or different kind of accident from any previously analyzed; or
3. involve a significant reduction in a margin of safety.

The determinations that the criteria set forth in 10 CFR 50.92 are met for this amendment request are indicated below:

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

   Response: No.

   The proposed license amendment would allow fuel assemblies to be removed from the reactor core and be stored in the Spent Fuel Pool in less time after subcriticality than currently allowed by the TSs. Decreasing the decay time of the fuel affects the isotopic make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated. The accident previously evaluated that is associated with the proposed license amendment is the fuel handling accident. Allowing the fuel to be offloaded as early as 100 hours after subcriticality does not impact the manner in which the fuel is offloaded. The accident initiator is the dropping of the fuel assembly. Since earlier offload does not effect fuel handling, there is no increase in the probability of occurrence of a fuel handling accident. The time frame in which the fuel assemblies are moved has been evaluated against the 10 CFR 50.67 dose limits for members of the public, licensee personnel and control room. Additionally, the guidance provided in Reg. Guide 1.183 was used for the selective application of Alternative Source Term. All dose limits are met with the reduced core offload times.

   During the period from October 15th through May 15th up to and including the year 2010, a fully radiated 193 element core can be off-loaded to a Spent Fuel Pool with a 100-hour in-vessel decay, rather than a 168 hour decay, because the Spent Fuel Pool Cooling System is capable of maintaining both pools below 180°F. The continued implementation of the Spent Fuel Pool Integrated Decay Heat Management Program provides the administrative controls required to maintain SFP temperatures below the 180°F limit.

   The accident previously evaluated that is associated with fuel movement is the Fuel Handling Accident. With this proposed amendment, the selected characteristics of the AST and the TEDE criteria become the design basis for the Fuel Handling Accident at Salem Units 1 and 2. Thus, there is no significant increase in consequences.

   Therefore, the proposed license amendment does not increase the probability of occurrence or the consequences of accidents previously evaluated are not increased.

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

   Response: No.

   The proposed license amendment would allow core offload to occur in less time after subcriticality, which affects the isotopic make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The isotopic makeup of the fuel assemblies and the amount of decay heat produced by the fuel assemblies do not currently initiate any accident. A change in the isotopic makeup of the fuel at the time of core offload or an increase in the decay heat produced by the fuel being offloaded will not cause the initiation of any accident. The accident previously evaluated that is associated with fuel movement is the fuel handling accident. There is no change to the manner in which fuel is being handled or in the equipment used to offload or store the fuel. The effects of the additional decay heat load have been analyzed. The analysis demonstrated that the existing Spent Fuel Pool cooling system and associated systems under worst-case circumstances would maintain the integrity of the Spent Fuel Pool. The proposed method of offload does not create a new or different kind of accident from any accident previously evaluated.

   Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The margin of safety pertinent to the proposed changes is the dose consequences resulting from a fuel handling accident. The shorter decay time prior to fuel movement has been evaluated against 10 CFR Part 50.67 and all limits continue to be met. In addition, the integrity of the Spent Fuel Pool has been demonstrated with the additional decay heat load. As stated above, the changes in isotopic makeup and additional heat load do not impact any safety settings and do not cause any safety limit to not be met. In addition, the integrity of the Spent Fuel Pool is maintained.

   The time frame in which the fuel assemblies are moved has been evaluated against the 10 CFR 50.67 dose limits for members of the public, licensee personnel and control room. Additionally, the guidance provided in Reg. Guide 1.183 was used for the selective application of Alternative Source Term. Calculations performed conclude that expected dose limits following a Fuel handling Accident are met with the proposed decay time prior to commencing fuel movement.

   During the period from October 15th through May 15th up to and including the year 2010, a fully radiated 193 element core can be off-loaded to a Spent Fuel Pool with a 100-hour in-vessel decay, rather than a 168 hour decay, because the Spent Fuel Pool Cooling System is capable of maintaining both pools below 180°F. The continued implementation of the Spent Fuel Pool Integrated Decay Heat Management Program provides the administrative controls required to maintain SFP temperatures below the 180°F limit.

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

Based on this review, it is concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, PSEG proposes that a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

The NRC's traditional methods for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the ASTs and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in other regulatory documents when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

PSEG used this regulatory guide extensively in the preparation of this "selective implementation". This application and the supporting analyses comply with this guidance as it applies to a Fuel Handling Accident.

Title 10, Code of Federal Regulations, Part 50 Section 67, "Accident Source Term".

10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design basis radiological consequences analyses. This document is part of a 10 CFR 50.90 license amendment application and evaluates the consequences of a design basis fuel handling accident as previously described in the Salem UFSAR.

USNRC Branch Technical Position ASB 9-2, Residual Decay Heat for Light-Water Reactors for Long-Term Cooling, Revision 2 of July 1981.

BTP ASB 9-2 uses a conservative approach for calculating fuel element decay heat, and is applied to this amendment without scaling factors or other adjustments.

Regulatory Guide 1.25, "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".

RG 1.183 supersedes corresponding radiological assumptions provided in other regulatory guides and standard review plan chapters when used in conjunction with an approved alternative source term and the TEDE provided in 10 CFR 50.67.

10 CFR 100, "Determination of Exclusion Area, Low Population Zone and Population Center Distance".

10 CFR 100.11 provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67. In accordance with the provisions of 10 CFR 50.67(a), PSEG applied the dose reference values in 10 CFR 50.67 (b) (2) in the analyses in lieu of 10 CFR 100 for the Fuel Handling Accident.

NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents".

The SRP Section 15.7.4 describes the radiological effects of a postulated Fuel Handling Accident. The SRP does not directly refer to the guidance of RG 1.183 or 10 CFR 50.67. Instead, it refers to regulatory documents, which are superseded by the selective application of the Alternative Source Term for the Fuel Handling Accident.

10 CFR 50 Appendix A, General Design Criteria 19, Control Room

PSEG has applied the guidelines provided by 10 CFR 50.67 and RG 1.183, which supersedes the current requirements of GDC 19 for the Fuel Handling Accident.

5.3 Conclusion

The FHA dose analyses were performed in accordance with AST and TEDE guidelines provided in Regulatory Guide 1.183 and 10 CFR 50.67. The assumptions and design inputs are listed in Engineering Calculations listed in the reference section. The SFP Cooling Capacity calculations were performed applying acceptable NRC guidance and conservatism aspects resulting in assurance that the design basis limits for SFP heat removal are maintained.

The results of these analyses indicate that the doses shown in pages 10 and 11 of this application are less than the TEDE criteria set forth in RG 1.183 and are an small fraction of the does criteria in 10CFR 50.67.

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.

Similar TS amendments were approved by the NRC as listed in Reference 7.10.

6.0 ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

PSEG has concluded that implementation of this amendment will have no adverse impact upon the Salem units; neither will it contribute to any significant additional quantity or type of effluent being available for adverse environmental impact or personnel exposure. The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite. The new consequences of the revised Fuel Handling Accident analysis remain well below the acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183.
It has been determined there is:

1. No significant hazards consideration,
2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and
3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Salem TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

7.0 REFERENCES

7.1 PSEG Calculation S-C-ZZ-MDC-1912, Revision 0
7.2 PSEG Calculation S-C-ZZ-MDC-1920, Revision 1
7.3 PSEG Calculation S-C-SF-MEE-1679, Revision 0
7.4 PSEG Salem Units 1 and 2, Final Safety Analysis Report
7.5 PSEG Salem Units 1 and 2, Technical Specifications
7.6 PSEG, Outage Risk Management, NC.OM.AP.ZZ-0001(Q)
7.7 10 CFR 50.67, "Accident Source Term"
7.8 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
7.9 NRC Branch Technical Position ASB 9-2 Revision 2 of July 1981, USNRC Standard Review Plan 9.2.5, Ultimate Heat Sink, NUREG 0800
7.10 American Electric Power, DC Cook Units 1 and 2, Amendment Nos. 260 and 243, dated November 30, 2001

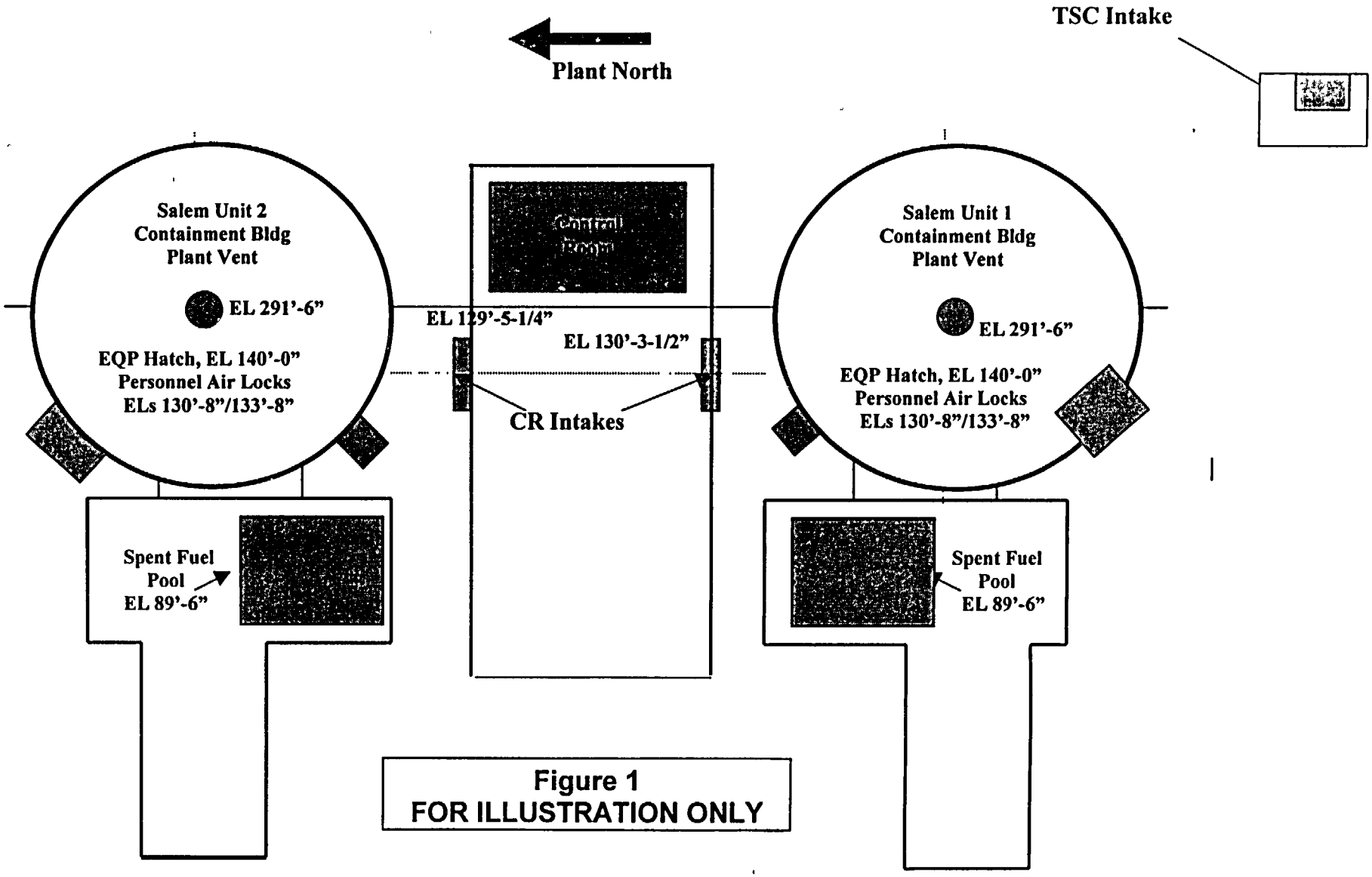

Figure 1
FOR ILLUSTRATION ONLY

Salem 1 & 2 Containment & Fuel Handling Building Orientation

SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
REVISIONS TO THE TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License DPR-70 are affected by this change request:

Technical Specification	Page
3/4.9.3	3/4 9-3
B 3/4.9.3	B 3/4 9-3

The following Technical Specifications for Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
3/4.9.3	3/4 9-3
B 3/4.9.3	B 3/4 9-3

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least: ~~168 hours~~.

a. **100 hours - Applicable through year 2010.**
b. **168 hours**

APPLICABILITY: ~~During movement of irradiated fuel in the reactor pressure vessel.~~
Specification 3.9.3.a - From October 15th through May 15th, during movement of irradiated fuel in the reactor pressure vessel.

Specification 3.9.3.b - From May 16th through October 14th, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than ~~168 hours~~ **the required time**, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical ~~for at least 168~~ **as required** by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

The sampling and analysis required by surveillance requirement 4.9.1.2 ensures the boron concentration required by Limiting Condition of Operation 3.9.1 is met. Sampling and analysis of the refueling canal is required if water exists in the refueling canal, regardless of the amount.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. **The 100-hour decay time is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in Reg. Guide 1.183.**

The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. Delaware River water average temperature between October 15th and May 15th is determined from historical data taken over 30 years. The use of 30 years of data to select maximum temperature is consistent with Reg. Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants".

A core offload has the potential to occur during both applicability time frames. In order not to exceed the analyzed Spent Fuel Pool cooling capability to maintain the water temperature below 180°F, two decay time limits are provided. In addition, PSEG has developed and implemented a Spent Fuel Pool Integrated Decay Heat Management Program as part of the Salem Outage Risk Assessment. This program requires a pre-outage assessment of the Spent Fuel Pool heat loads and heatup rates to assure available Spent Fuel Pool cooling capability prior to offloading fuel.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least ~~168 hours~~:

a. **100 hours - Applicable through year 2010.**
b. **168 hours**

APPLICABILITY: ~~During movement of irradiated fuel in the reactor pressure vessel.~~
Specification 3.9.3.a - From October 15th through May 15th, during movement of irradiated fuel in the reactor pressure vessel.

Specification 3.9.3.b - From May 16th through October 14th, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than ~~168 hours~~ **the required time**, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical ~~for at least 168 hours~~ **as required** by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

The sampling and analysis required by surveillance requirement 4.9.1.2 ensures the boron concentration required by Limiting Condition of Operation 3.9.1 is met. Sampling and analysis of the refueling canal is required if water exists in the refueling canal, regardless of the amount.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. **The 100-hour decay time is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in Reg. Guide 1.183.**

The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. Delaware River water average temperature between October 15th and May 15th is determined from historical data taken over 30 years. The use of 30 years of data to select maximum temperature is consistent with Reg. Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants".

A core offload has the potential to occur during both applicability time frames. In order not to exceed the analyzed Spent Fuel Pool cooling capability to maintain the water temperature below 180°F, two decay time limits are provided. In addition, PSEG has developed and implemented a Spent Fuel Pool Integrated Decay Heat Management Program as part of the Salem Outage Risk Assessment. This program requires a pre-outage assessment of the Spent Fuel Pool heat loads and heatup rates to assure available Spent Fuel Pool cooling capability prior to offloading fuel.