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PSEG
Nuclear L.L.C.

10 CFR 50.90

APR 0 9 2009

LR-N09-0034
LAR S08-06

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

Salem Generating Station, Units 1 and 2
Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: **License Amendment Request to Relocate Communications, Manipulator Crane, and Crane Travel Requirements from Technical Specifications**

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) requests a change to the Technical Specifications (TS) for Salem Generating Station (SGS) Units 1 and 2. The proposed change would relocate three TS requirements pertaining to communications during refueling operations (TS 3/4.9.5), manipulator crane operability (TS 3/4.9.6), and crane travel (TS 3/4.9.7) to the SGS Technical Requirements Manual (TRM).

The relocation of the three TS is justified because they do not meet the criteria of 10 CFR 50.36(c)(2)(ii), "Limiting Condition for Operation," for retention in the TS. The proposed relocations are also consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3, dated June 2004, as that guidance does not contain requirements for communications, manipulator crane operability, or crane travel. The Limiting Conditions for Operation (LCOs), Applicability, Actions, and Surveillance Requirements for both units will be retained in the TRM. Future changes to TRM requirements will be controlled under 10 CFR 50.59.

Attachment 1 to this letter describes the proposed changes and provides the justification for the changes. PSEG has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92. Attachment 2 contains the marked-up TS pages. Attachment 3 contains, for information only, the marked up TS Bases pages. Attachments 2 and 3 include documents for both SGS Units 1 and 2. No regulatory commitments are contained in this submittal.

Following approval of this license amendment request, PSEG intends to revise the TRM under the provisions of 10 CFR 50.59 to permit the necessary suspension of certain heavy loads over irradiated fuel for a short time during dry cask loading operations in the fuel transfer pool. Specifically, the spent fuel canister lid, which weighs in excess of the 2200 pound limit currently in LCO 3.9.7 will need to be suspended over the fuel-filled canister during lid installation. The lid will be lifted only with a single failure proof lifting system meeting the guidance in NUREG-0612. To support the 2010 SGS dry cask storage campaign, PSEG requests approval of the proposed amendment by April 30, 2010, with implementation to be completed within 60 days.

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In accordance with 10 CFR 50.91(b)(1), a copy of this submittal is being sent to the State of New Jersey by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Mr. Jeffrie Keenan at (856) 339-5429

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

Executed on 4/8/09
(date)

Sincerely,

Robert C. Braun
Site Vice President
Salem Generating Station

RCB/bg

Attachments: (3)

cc: S. Collins, Regional Administrator – NRC Region I
R. Ennis, Project Manager - Salem, USNRC
NRC Senior Resident Inspector - Salem
P. Mulligan, Manager IV, NJBNE
Commitment Coordinator – Salem
PSEG Corporate Commitment Manager

ATTACHMENT 1

License Amendment Request

Salem Generating Station, Units 1 and 2

Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Description of Proposed Changes, Technical Analysis, and Regulatory Analysis

Subject: Relocation of Communications, Manipulator Crane, and Crane Travel From Technical Specifications to the Technical Requirements Manual

ATTACHMENT 1
DESCRIPTION OF PROPOSED CHANGES, TECHNICAL ANALYSIS, AND REGULATORY ANALYSIS

1.0 DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) requests a change to the Technical Specifications (TS) for Salem Generating Station (SGS) Units 1 and 2. The proposed change would relocate three TS requirements pertaining to communications during refueling operations (TS 3/4.9.5), manipulator crane operability (TS 3/4.9.6), and crane travel (TS 3/4.9.7) to the SGS Technical Requirements Manual (TRM).

The relocation of the three TS is justified because they do not meet the criteria of 10 CFR 50.36(c)(2)(ii), "Limiting Condition for Operation," for retention in the TS. The proposed relocations are also consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3, dated June 2004, as that guidance does not contain requirements for communications, manipulator crane operability, or crane travel. The Limiting Conditions for Operation (LCOs), Applicability, Actions, and Surveillance Requirements in the relocated TS for both units will be retained in the TRM. Future changes to TRM requirements will be controlled under 10 CFR 50.59.

2.0 PROPOSED CHANGES

The proposed amendment would relocate current TS 3/4.9.5, 3/4.9.6, and 3/4.9.7, which provide requirements associated with control room communications during refueling, operability of the manipulator crane during movement of control rods or fuel assemblies within the reactor pressure vessel, and the control of heavy loads over fuel assemblies in the fuel storage pool, respectively, to the TRM. Markups to the TS pages are contained in Attachment 2 and markups of the affected TS Bases pages are contained in Attachment 3.

2.1 TS 3/4.9.5: Communications

LCO 3.9.5 currently states:

> 3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

This LCO is applicable during CORE ALTERATIONS. The proposed amendment would relocate TS 3/4.9.5 (LCO, APPLICABILITY, ACTION, and SURVEILLANCE) to the TRM with no changes. The TS page will accordingly be marked:

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2.2 TS 3/4.9.6: Manipulator Crane Operability

LCO 3.9.6 currently states:

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and

2. An overload cut off limit less than or equal to 2850 pounds

b. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds

This LCO is applicable during movement of control rods or fuel assemblies within the reactor pressure vessel. The proposed amendment would relocate TS 3/4.9.6 (LCO, APPLICABILITY, ACTION, and SURVEILLANCE) to the TRM with no changes. The TS page will accordingly be marked:

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2.3 TS 3/4.9.7: Crane Travel – Fuel Handling Area

LCO 3.9.7 currently states:

3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

This LCO is applicable with fuel assemblies in the storage pool. PSEG would also conservatively assume this LCO to be applicable with fuel assemblies in the fuel transfer pool during spent fuel cask loading operations. The proposed amendment would relocate TS 3/4.9.7 (LCO, APPLICABILITY, ACTION, and SURVEILLANCE) to the TRM with no changes. The TS page will accordingly be marked:

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After relocation of TS 3/4.9.7 to the TRM, PSEG intends to revise the requirements under the provisions of 10 CFR 50.59 to support dry spent fuel cask loading operations in the transfer pool. While no long-term storage of spent fuel takes place in the fuel transfer pool, dry storage cask loading operations are conducted there, which involves handling a heavy load over the irradiated fuel in the canister. The LCO will be changed to permit the handling of heavy loads that are necessary for dry spent fuel cask operations and exceed 2200 pounds (i.e., the canister lid) to be suspended over the irradiated fuel in the spent fuel canister provided that those loads are handled with a single-failure-proof lifting system meeting the guidance in NUREG-0612, Section 5.1.6.

In summary, the proposed amendment would relocate TS 3/4.9.5, 3/4.9.6 and 3/4.9.7 to the TRM, verbatim, and LCO 3.9.7 will subsequently be revised under 10 CFR 50.59 as necessary to permit dry spent fuel cask loading operations to be conducted as required by the dry storage system certificate of compliance and FSAR.

The following associated changes will also be made to support relocation of TS 3/4.9.5, 3/4.9.6, 3/4.9.7:

1. The TS Index will be revised to delete the references to TS 3/4.9.5, 3/4.9.6, and 3/4.9.7.

2. TS Bases Sections 3/4.9.5, "Communications," 3/4.9.6, "Manipulator Crane," and 3/4.9.7, "Crane Travel-Spent Fuel Storage Building," will be deleted.

Marked up TS pages are provided in Attachment 2. Marked-up TS Base pages are provided, for information only, in Attachment 3.

3.0 BACKGROUND

10 CFR 50.36 (Reference 1) specifies four criteria for including LCOs in the TS for commercial nuclear power reactors. According to 10 CFR 50.36(c)(2)(ii), an LCO must be established for items that meet one or more of the criteria. None of the three LCOs proposed for relocation meet any of the four criteria. Furthermore, all three of these LCOs were excluded from the improved standard technical specifications for Westinghouse nuclear plants described in NUREG-1431 (Reference 2). The applicability of each of the four criteria of §50.36 to each of the TS proposed to be relocated to the TRM is discussed in Section 4.0 below.

The changes proposed in this submittal are consistent with changes previously approved in Amendments 190 and 134 for St. Lucie Units 1 and 2 (Reference 5), and in Amendment 200 for Kewaunee Power Station (Reference 6).

4.0 TECHNICAL ANALYSIS

10 CFR 50.36(c)(2)(ii) specifies the four criteria for including LCOs in the TS for commercial nuclear power reactors:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary (RCPB).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

In June 2004, the NRC issued Revision 3 to NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Relocation of the three TS described in this submittal is consistent with NUREG-1431. The evaluation of each of the TS proposed for relocation is provided below:

4.1 TS 3/4.9.5: Communications

LCO 3.9.5 is applicable only during CORE ALTERATIONS, which can only be conducted with the reactor head removed and the reactor coolant system depressurized. The components covered by this LCO include radios and associated power and transmission equipment necessary to establish and maintain communications between the control room and the refueling station.

§50.36(c)(2)(ii)(A), [Criterion 1], applies to installed instrumentation that is used to detect, and indicate in the control room, significant abnormal degradation of the RCPB. Equipment used by personnel to establish and maintain communication between the control room and the refueling station is not part of any installed instrumentation that performs the design functions described in this criterion. Therefore, Criterion 1 is not met for this LCO.

Criterion 2 applies to a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Equipment used by personnel to establish and maintain communication between the control room and the refueling station is not a process variable, design feature, or operating restriction as described in this criterion. Therefore, Criterion 2 is not met for this LCO.

Criterion 3 applies to a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Equipment used by personnel to establish and maintain communication between the control room and the refueling station does not include any systems, structures, or components that perform the design functions described in this criterion. Therefore, Criterion 3 is not met for this LCO.

Criterion 4 applies to a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Equipment used by personnel to establish and maintain communication between the control room and the refueling station has not been shown in any operating experience or probabilistic safety assessment to be significant to public health and safety. Therefore, Criterion 4 is not met for this LCO.

Because TS 3/4.9.5 meets none of the §50.36(c)(2)(ii) criteria for retention as an LCO in the TS, it may be relocated to the TRM.

4.2 TS 3/4.9.6: Manipulator Crane Operability

LCO 3.9.6 is applicable only during movement of control rods or fuel assemblies within the reactor vessel, which can only take place with the reactor head removed and the reactor coolant system depressurized. This LCO governs the manipulator crane and the auxiliary hoist located inside the Reactor Building. The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the reactor cavity and runs on rails set into the floor along the edge of the reactor cavity. The bridge and trolley motions are used during refueling operations with the reactor head removed to position the vertical mast over a fuel assembly in the reactor core and move the mast as necessary for refueling the reactor.

Inside the manipulator crane, a long tube with a pneumatic gripper on the end is lowered down from the mast to grip fuel assemblies. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel, while inside the mast tube, is transported to its new position. The manipulator crane is used to move individual fuel assemblies with or without inserts (e.g., integral rod cluster control assembly (RCCAs), wet annular burnable absorbers (WABAs), etc.) into, out of, and between positions in the reactor core during refueling operations. The manipulator crane is limited to moving a single fuel assembly at a time by its physical configuration and lift capacity limits. An overload cutoff device in the manipulator crane prevents it from applying excessive lifting force in the event it is inadvertently engaged with an object other than a fuel assembly during lifting operations.

The auxiliary hoist is currently only used to facilitate latching and unlatching the control rod drives for individual RCCAs in the reactor core during refueling operations. The auxiliary hoist is not used to lift and move fuel assemblies.

§50.36(c)(2)(ii)(A), [Criterion 1] applies to installed instrumentation that is used to detect, and indicate in the control room, significant abnormal degradation of the RCPB. The manipulator crane and auxiliary hoist do not meet this criterion because they are not related to any installed instrumentation that performs these functions. Therefore, Criterion 1 is not met for this LCO.

Criterion 2 applies to a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The initial conditions of Criterion 2 are not limited to only process variables assumed in safety analyses, but also include certain active design features and operating restrictions needed to preclude unanalyzed accidents. Active design features are intended to be those design features under the control of operations personnel (i.e., licensed operators and personnel who perform control functions at the direction of licensed operators). Should an LCO involve a physical, designed-in plant feature that prevents operations staff from immediately placing the plant in an unanalyzed condition in the course of operations (one that would require a design change before operators could exceed the limits of the LCO) that LCO would not satisfy Criterion 2. Two design basis accidents have been identified that could involve the manipulator crane:

1. UFSAR Section 15.3.3: Inadvertent Loading of a Fuel Assembly into an Improper Position
2. UFSAR Section 15.4.6: Fuel Handling Accident (FHA)

Each of the above accidents is discussed individually below as it relates to Criterion 2 for the manipulator crane LCO.

<u>Inadvertent Loading of a Fuel Assembly into an Improper Position</u>

The manipulator crane would have been used to load a fuel assembly into an improper position if that event were to occur. However, LCO 3.9.6 only provides limits on the minimum load capacities and load limit controls required for the manipulator crane and auxiliary hoist. This LCO has no bearing on the process used to ensure fuel assemblies are moved into the proper position in the core; thus LCO 3.9.6 does not prevent the misloading of a fuel assembly or otherwise involve the initiating conditions for this accident. Therefore, Criterion 2 is not met by this LCO for the Inadvertent Loading of Fuel Assembly into an Improper Position event.

Fuel Handling Accident (FHA)

As described in SGS UFSAR Section 15.4.6, an FHA involves the drop of a single irradiated fuel assembly and can occur in either of two locations: inside containment or in the Fuel Handling Building (FHB). For an FHA postulated to occur in containment, the fuel assembly would be dropped from the manipulator crane. The manipulator crane cannot physically access the FHB, so it plays no role in the FHA postulated to occur there. LCO 3.9.6 only provides a minimum load capacity for the manipulator crane and auxiliary hoist, and requirements for load limiting devices for the crane and hoist. The fuel assembly, the manipulator crane, and the auxiliary hoist all have physical, designed-in features that would prevent the operators from inadvertently placing the plant in an unanalyzed condition. In order to use a different fuel assembly in the reactor or a different manipulator crane or auxiliary hoist that could change an initial condition for the FHA, a design change and review under the provisions of 10 CFR 50.59 would be required. Therefore, Criterion 2 is not met by this LCO for the Fuel Handling Accident.

Criterion 3 applies to a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The manipulator crane and auxiliary hoist are used solely during refueling operations with the reactor head removed. They do not meet this criterion because they do not actuate to mitigate a design basis accident or transient analysis and are not related to equipment that does. Therefore, Criterion 3 is not met for this LCO.

Criterion 4 applies to a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The manipulator crane and auxiliary hoist have not been shown in any operating experience or probabilistic safety assessment to be significant to public health and safety. Therefore, Criterion 4 is not met for this LCO.

Because TS 3/4.9.6 meets none of the §50.36(c)(2)(ii) criteria for retention as an LCO in the TS, it may be relocated to the TRM.

4.3 TS 3/4.9.7: Crane Travel – Fuel Handling Area

LCO 3.9.7 is applicable whenever fuel assemblies are in the storage pool (i.e., at all times).This LCO limits the weight of any load carried over fuel in the storage pool to 2200 pounds. PSEG would also consider this LCO to be applicable when handling heavy loads over fuel in the fuel transfer pool when loading a spent fuel cask. The 2200 pound weight is the nominal combined weight of a single fuel assembly, an RCCA, and a fuel handling tool (Reference TS Bases 3/4.9.7). But, the 2200 pound weight is not used to determine the amount of radioactivity released as a result of the event. The LCO is used (indirectly) to preclude handling more than one fuel assembly over irradiated fuel at a time with a non-single-failure-proof lifting system. See additional discussion under Criterion 2 below.

§50.36(c)(2)(ii)(A), [Criterion 1] applies to installed instrumentation that is used to detect, and indicate in the control room, significant abnormal degradation of the RCPB. The 2200 pound limit for loads suspended over irradiated fuel does not meet this criterion because it is a weight limit for lifts in certain areas of the plant and is not related to any installed instrumentation that performs these functions. Therefore, Criterion 1 is not met for this LCO.

Criterion 2 applies to a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The initial conditions of Criterion 2 are not limited to only process variables assumed in safety analyses, but also include certain active design features and operating restrictions needed to preclude unanalyzed accidents. Active design features are intended to be those design features under the control of operations personnel (i.e., licensed operators and personnel who perform control functions at the direction of licensed operators). Should an LCO involve a physical, designed-in plant feature that prevents operations staff from immediately placing the plant in an unanalyzed condition in the course of operations (one that would require a design change before operators could exceed the limits of the LCO) that LCO would not satisfy Criterion 2. The only design basis accident of interest for this LCO is the Fuel Handling Accident (FHA). The relationship of this LCO and Criterion 2 for the FHA is discussed below.

<u>Fuel Handling Accident Criticality Consequences</u>

The spent fuel racks in the fuel storage pool are designed to withstand the impact of the drop of a single fuel assembly (SGS UFSAR Section 9.1.2.3). The rack design prevents a critical array as a result of an FHA.

<u>Fuel Handling Accident Radiological Consequences</u>

The radiological consequences of a design basis FHA are described in Section 15.4.6 of the SGS UFSAR. The radiological consequences of the FHA were determined in accordance with the guidance in Regulatory Guide 1.183 (Reference 3), assuming all 264 fuel rods in a peak power fuel assembly are breached as a result of the drop. The failure of all fuel rods is a conservative assumption that provides a bounding source term for determining the potential radiological consequences resulting from any level of damage to a single fuel assembly. Thus, there is no mechanistic relationship between the weight of 2200 pounds established by this LCO and the fuel damage assumed in the FHA radiological analysis.

An FHA involves the drop of a single irradiated fuel assembly and can occur in either of two locations: inside containment and in the FHB. The FHA postulated to occur in containment would not involve the storage pool as addressed in this LCO because the storage pool is not located in containment. The fuel transfer pool is also not located inside containment. Thus, the FHA inside containment is not applicable to this LCO. The FHA postulated to occur in the FHB involves the drop of a single irradiated assembly from the fuel handling crane during movement in and around the spent fuel pool or fuel transfer pool.

The two cranes capable of carrying heavy loads over the storage pool or fuel transfer pool are the fuel handling crane and the cask handling crane. Each crane is discussed separately below.

Fuel Handling Crane

Individual fuel assemblies are handled in the fuel storage pool or fuel transfer pool by means of a special tool suspended from the fuel handling crane hook. The fuel handling crane, by design, can only handle a single fuel assembly and integral RCCA at one time. Both the fuel assembly and the fuel handling crane have physical, designed-in features that would prevent the operators from inadvertently placing the plant in an unanalyzed condition. In order to use a different fuel assembly in the reactor or to use a crane that could lift more than one assembly (i.e., modifications that could change an initial condition for the FHA), a design change and review under the provisions of 10 CFR 50.59 would be required. Thus, Criterion 2 is not met by this LCO for the fuel handling crane.

Cask Handling Crane

The SGS cask handling crane is a bridge and trolley crane used to: 1) transfer new fuel containers from the truck bay to the laydown area near the new fuel storage area, and 2) move spent fuel casks from the transfer pool to the decontamination pit and to the truck bay. The cask handling crane bridge rails are located such that the cask handling crane can only travel over the fuel transfer pool, decontamination pit, new fuel handling area, and receiving bay (see UFSAR Figure 9.1-1). It cannot carry loads over the spent fuel pool where long-term storage of irradiated fuel takes place.

The physical arrangement of the FHB is such that the transfer pool in each unit is separated from the spent fuel pool by a full-height, solid concrete wall with one 32-inch wide gate canal to permit movement of single fuel assemblies between the pools. The gate canal is designed to permit isolation of the two pools from each other by the manual installation of two sluice gates arranged in series. The redundant sluice gates are installed so that the transfer pool may be drained for maintenance on the fuel transfer equipment without draining the spent fuel pool. Normally, the fuel transfer pool is water-filled and the sluice gates are not installed. Water loss from the spent fuel pool due to the failure or accidental opening of a single sluice gate when the transfer pool is empty (or level lowered) will not occur due to the redundancy in the sluice gates. The bottom elevation of the gate canal in the wall is approximately 104 ft. The elevation of the top of the spent fuel in the wet storage racks in the spent fuel pool is approximately 104 ft. Thus, an event postulated to occur in the fuel transfer pool when the sluice gates are not installed that results in a complete draining of the fuel transfer pool would, at worst, drain the spent fuel pool to an elevation approximately equal to the top of the stored fuel[1].

A spent fuel cask or any other load lifted by the cask handling crane can travel only over the fuel transfer pool. The integrity of the spent fuel pool will not be breached due to a heavy load drop in the fuel transfer pool since each of the pool structures are separate and distinct (SGS UFSAR Section 9.1.3.3). For this reason, the SGS licensing basis does not include an analysis of cask drop in the transfer pool (SGS UFSAR Section 9.1.4.3).

[1] Heavy load handling in the fuel transfer pool during future spent fuel dry storage cask loading will be performed using the upgraded, single-failure-proof cask handling crane.

Regulatory Guide 1.13, Revision 2 (Reference 4) provides NRC's current guidance pertaining to spent fuel storage facility design. Regulatory Position C.5 in RG 1.13 addresses the control of heavy loads and suggests one of the following two conditions should be met:

a) Cranes should be designed to provide single-failure-proof handling of heavy loads, so that a single failure will not result in the crane handling system losing the capability to perform its safety function.

b) The spent fuel cask-loading area should be designed to withstand, without significant leakage of the adjacent spent fuel storage [area], the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

SGS was licensed based on Revision 0 to this Regulatory Guide (Safety Guide 1.13 at the time). Regulatory Position C.5 of Safety Guide 1.13 recommended that one of the following heavy load control provisions for spent fuel facility design be met:

a) Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving into the vicinity of the pool, or

b) The fuel pool should be designed to withstand, without leakage which would uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted. If this latter approach is followed, design provisions should be made to prevent this crane, when carrying heavy loads, from moving in the vicinity of stored fuel.

The SGS fuel transfer pool is robustly constructed and does not contain spent fuel assemblies stored in spent fuel racks. It is used to stage new fuel for reactor refueling and to load spent fuel casks. The gate canal opening in the wall separating the spent fuel pool and the fuel transfer pool permits the movement of single fuel assemblies between the two pools. Long-term storage of irradiated fuel takes place only in the spent fuel storage pool, not in the fuel transfer pool. The cask handling crane can only travel and carry loads over the fuel transfer pool due by design. Heavy load handling incidents occurring in the fuel transfer pool with the redundant sluice gates removed will not cause the fuel in the spent fuel pool to become uncovered because the elevation of the top of the spent fuel in the spent fuel racks is the same as the elevation of the bottom of the gate canal in the adjacent wall, as discussed above. Therefore, the design of the SGS spent fuel storage facility meets both Conditions 'a' and 'b' of Regulatory Position C.5 in Safety Guide 1.13. In addition, the intent of Condition 'b' of Regulatory Position C.5 in Regulatory Guide 1.13, Revision 2 is met.

In summary, Criterion 2 is not met by this LCO for the Fuel Handling Accident.

Criterion 3 applies to a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The 2200 pound limit for loads suspended over irradiated fuel does not meet this criterion because it is a weight limit for lifts in certain areas of the plant and it is not related to any equipment that functions or actuates to mitigate a design basis accident or transient analysis. Therefore, Criterion 3 is not met for this LCO.

Criterion 4 applies to a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The 2200 pound limit for loads suspended over irradiated fuel has not been shown in any operating experience or probabilistic safety assessment to be significant to public health and safety. Therefore, Criterion 4 is not met for this LCO.

Because TS 3/4.9.7 meets none of the §50.36(c)(2)(ii) criteria for retention as an LCO in the TS, and meets the intent of Regulatory Guide 1.13, Position C.5, it may be relocated to the TRM.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No.

The proposed relocation of Technical Specifications 3/4.9.5, 3/4.9.6, and 3/4.9.7 does not alter the requirements for component operability or surveillance currently in the Technical Specifications. The proposed change to remove these requirements from the Technical Specifications and relocate the information to an administratively controlled document will have no impact on any safety related structures, systems or components.

The probability of occurrence of a previously evaluated accident is not increased because this change does not introduce any new potential accident initiating conditions. The consequences of accidents previously evaluated in the UFSAR are not affected because the ability of the components to perform their required function is not affected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes are fundamentally administrative in nature because they do not result in physical alterations or changes in the method by which any safety related system performs its intended function. The proposed changes do not affect any safety analysis assumptions. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

All requirements will continue to be implemented as they are now as defined in the Technical Specifications. The proposed revision does not make changes in any method of testing or how any safety related system performs its safety functions.

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

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

Response: No.

The proposed change to remove Technical Specifications 3/4.9.5, 3/4.9.6, and 3/4.9.7 from the Technical Specifications and relocate them to the Technical Requirements Manual does not alter the requirements for operability of the affected equipment in TS 3/4.9.5 and 3/4.9.6, or compliance with the weight limit of TS 3/4.9.7. Future revisions to the Technical Requirements Manual will be subject to review pursuant to 10 CFR 50.59. The proposed change will not affect the current Technical Specification requirements or the components to which they apply.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PSEG concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 FR 50.36 establishes four criteria for having a Limiting Condition for Operation (LCO) in the plant Technical Specifications. Meeting at least one of those criteria requires an LCO to be established. The TS proposed for relocation meet not of the four criteria as explained above.

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.

6.0 ENVIRONMENTAL CONSIDERATION

A review 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.

7.0 REFERENCES

1. U.S. Code of Federal Regulations, Title 10, “Energy,” Part 50, “Production and Utilization Facilities,” §50.36, “Technical Specifications.”

2. NUREG-1431, “Standard Technical Specifications for Westinghouse Plants,” Volume 1, Revision 3.0, June 2004.

3. Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

4. USNRC Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” Revision 2.

5. NRC Safety Evaluation Related to Amendment Nos. 190 and 134, St. Lucie Units 1 and 2 (TAC Nos. MB5667 and MB5668), dated April 28, 2004.

6. NRC Safety Evaluation Related to Amendment No. 200, Kewaunee Power Station (TAC No. MD7300), November 20, 2008.

ATTACHMENT 2

Salem Generating Station

Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Relocation of Communications, Manipulator Crane, and Crane Travel Requirements from 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
Table of Contents	IX
3/4.9.5	3/4.9-5
3/4.9.6	3/4.9-6
3/4.9.7	3/4.9-7

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

Technical Specification	Page
Table of Contents	IX
3/4.9.5	3/4.9-5
3/4.9.6	3/4.9-6
3/4.9.7	3/4.9-7

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~COMMUNICATIONS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.5 Direct communication shall be maintained between the control room and personnel at the refueling station.~~

~~APPLICABILITY: During CORE ALTERATIONS.~~

~~ACTION:~~

~~When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.~~

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~~MANIPULATOR CRANE OPERABILITY~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:~~

~~a. The manipulator crane used for movement of fuel assemblies having:~~

~~1. A minimum capacity of 3250 pounds, and~~

~~2. An overload cut off limit less than or equal to 2850 pounds.~~

~~b. The auxiliary hoist used for movement of control rods having:~~

~~1. A minimum capacity of 700 pounds, and~~

~~2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.~~

~~APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.~~

~~ACTION:~~

~~With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off set at less than or equal to 2850 pounds; this includes the heavy load plus the weight of the crane mast and gripper.~~

~~4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.~~

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~~CRANE TRAVEL - FUEL HANDLING AREA~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.~~

~~APPLICABILITY: With fuel assemblies in the storage pool.~~

~~ACTION:~~

~~With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.7 The overload cutoff which prevents crane travel with loads in excess of 2200 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during the crane operation.~~

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INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REFUELING OPERATIONS

~~3/4.9.5 COMMUNICATIONS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.~~

~~APPLICABILITY: During CORE ALTERATIONS.~~

~~ACTION:~~

~~When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.~~

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~~MANIPULATOR CRANE OPERABILITY~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:~~

~~a. The manipulator crane used for movement of fuel assemblies having:~~

~~1. A minimum capacity of 3250 pounds, and~~

~~2. An overload cut off limit less than or equal to 2850 pounds.~~

~~b. The auxiliary hoist used for movement of control rods having:~~

~~1. A minimum capacity of 700 pounds, and~~

~~2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.~~

~~APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.~~

~~ACTION:~~

~~With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off set at less than or equal to 2850 pounds; this includes the heavy load plus the weight of the crane mast and gripper.~~

~~4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.~~

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~~CRANE TRAVEL - FUEL HANDLING AREA~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.~~

~~APPLICABILITY: With fuel assemblies in the storage pool.~~

~~ACTION:~~

~~With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.7 The overload cutoff which prevents crane travel with loads in excess of 2200 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during the crane operation.~~

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ATTACHMENT 3

Salem Generating Station

Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Relocation of Communications, Manipulator Crane, and Crane Travel Requirements from Technical Specifications

<u>TECHNICAL SPECIFICATION BASES PAGES WITH PROPOSED CHANGES</u>

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

Technical Specification	Page
B 3/4.9.5	B 3/4.9-3
B 3/4.9.6	B 3/4.9-3
B 3/4.9.7	B 3/4.9-3

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

Technical Specification	Page
B 3/4.9.5	B 3/4.9-3
B 3/4.9.6	B 3/4.9-3
B 3/4.9.7	B 3/4.9-3

The surveillance requirement 4.9.4.2 demonstrates that the necessary hardware, tools, and equipment are available to close the equipment hatch. The surveillance is performed prior to movement of irradiated fuel assemblies within the containment. This surveillance is only required to be met when the equipment hatch is to be open during fuel movement.

3/4.9.5 COMMUNICATIONS

Deleted. ~~The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.~~

3/4.9.6 MANIPULATOR CRANE

Deleted. ~~The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

Deleted. ~~The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.~~

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirements that at least one residual heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. A minimum flow rate of 1000 gpm is required. Additional flow limitations are specified in plant procedures, with the design basis documented in the Salem UFSAR. These flow limitations address the concerns related to vortexing and air entrapment in the Residual Heat Removal system, and provide operational flexibility by adjusting the flow limitations based on time after shutdown. The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability.

The surveillance requirement 4.9.4.2 demonstrates that the necessary hardware, tools, and equipment are available to close the equipment hatch. The surveillance is performed once per refueling prior to the start of movement of irradiated fuel assemblies within the containment. This surveillance is only required to be met when the equipment hatch is open.

3/4.9.5 COMMUNICATIONS

Deleted. ~~The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.~~

3/4.9.6 MANIPULATOR CRANE

Deleted. ~~The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

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The requirements that at least one residual heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. A minimum flow rate of 1000 gpm is required. Additional flow limitations are specified in plant procedures, with the design basis documented in the Salem UFSAR. These flow limitations address the concerns related to vortexing and air entrapment in the Residual Heat Removal system, and provide operational flexibility by adjusting the flow limitations based on time after shutdown. The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability.

For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that the two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
  A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disable.