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Mark Morgan
Regulatory Affairs

June 5, 2017

10 CFR 50.59
10 CFR 50.71
10 CFR 72.48

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

Subject: **Docket Nos. 50-206, 50-361, 50-362, and 72-41**
Facility Change Report, Summary Report of Commitment Changes, and Cycle Specific Technical Specification Bases Page Updates
San Onofre Nuclear Generating Station (SONGS) Units 1, 2, 3, and the Independent Spent Fuel Storage Installation

Dear Sir or Madam:

The attached enclosures contain the Facility Change Report required by 10 CFR 50.59(d)(2) for SONGS Units 1, 2 and 3, and by 10 CFR 72.48(d)(2) for the SONGS ISFSI during the reporting period from April 1, 2015 through April 1, 2017. The enclosed reports provide a brief description and summary of the 10 CFR 50.59 evaluations performed for any changes, tests, and experiments for SONGS Units 2 and 3 (Enclosure 1) and a brief summary of any 10 CFR 72.48 evaluations performed for the SONGS ISFSI (Enclosure 2). There were no 10 CFR 50.59 evaluations performed for SONGS Unit 1 during this time period. Complete change documentation for the SONGS Units 2 and 3 10 CFR 50.59 evaluations and 10 CFR 72.48 evaluations performed for the SONGS ISFSI is available onsite.

The letter also provides the report of a summary of commitment changes following the guidance of Nuclear Energy Institute (NEI) 99-04, "Guidance for Managing NRC Commitment Changes'" Revision 0. For the reporting period of April 1, 2015 through April 1, 2017, no commitments were identified as reportable under the NEI guidelines

As required under SONGS Units 2 and 3 Technical Specification (TS) 5.4.4, changes to the SONGS Units 2 and 3 TS Bases made without prior Nuclear Regulatory Commission (NRC) approval are provided to the NRC on a frequency consistent with 10 CFR 50.71(e)(Enclosure 2). The reporting period for the changes to the SONGS Units 2 and 3 TS Bases is from April 1, 2015 through April 1, 2017. Only one TS Bases change package was processed during this time frame, which implemented the Permanently Defueled Technical Specifications. Because this change was a complete replacement of the previous SONGS Units 2 and 3 TS Bases, a copy of the current Bases is provided in Enclosure 2 as the change report.

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NMSS26
NRR
NMSS

P.O. Box 128
San Clemente, CA 92672

This letter or the attached enclosures do not contain any new commitments.

Should you have any questions or require additional information, please contact me at (949) 368-6745.

Sincerely,

Enclosures: 1. SONGS Units 2 and 3 10 CFR 50.59 Evaluation Summaries
2. SONGS ISFSI 10 CFR 72.48 Evaluations Summaries
3. SONGS Units 2 and 3 Technical Specification Bases Changes

cc: K. Kennedy, Regional Administrator, NRC Region IV
M. G. Vaaler, NRC Project Manager, San Onofre Unit s 1, 2 and 3
W. C. Allen, NRC Project Manager, SONGS ISFSI

ENCLOSURE 1

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

FACILITY CHANGE REPORT (FCR)

10 CFR 50.59 EVALUATION SUMMARIES

FOR THE PERIOD

FROM APRIL 1, 2015 THROUGH April 1, 2017

10CFR50.59 Evaluation 801288535-0120, Removal from Service of the Offsite Probable Maximum Flood (PMF) Berm and Channel

Description:
The proposed activity permanently "removes from service" the Offsite Probable Maximum Flood (PMF) Berm and Channel in accordance with the Decommissioning (DEC) process. NECP 801288535 revises UFSAR Section 02.04 "Hydrologic Engineering" to document that SONGS Offsite PMF Berm and Channel no longer perform a design function and are therefore "Removed from Service." NECP 801288535 does not physically modify the PMF Berm and Channel. However, maintenance and inspection requirements for the structure will be suspended.

Evaluation Summary:

The evaluation determined

(1) Of the credible defueled condition events, only the Spent Fuel Pool (SFP) Boiling Accident is relevant to removing the Offsite Probable Maximum Flood (PMF) Berm and Channel from service. The incremental increase in maximum water (flood) levels at SONGS 2&3 caused by the postulated removal of the Offsite PMF Berm are within the existing maximum flood elevations described in the UFSAR. The small increase in the volume of water that would pass under non-watertight doors does not result in more than a minimal increase in the probability of loss of SFP cooling.

(2) Any small addition to the volume of water passing under non-watertight doors would flow to low elevation collection areas (e.g., tunnels, vaults), is described in the existing analyses, and would not represent more than a minimal increase in the likelihood of occurrence of a malfunction or shutdown of the SFP cooling pumps.

(3) The postulated removal of the Offsite PMF Berm results in a small increase in the maximum water (flood) levels at SONGS 2&3. This increase in water level does not impact any structures, systems, and components that might increase the consequences of the Spent Fuel Pool (SFP) Boiling Accident. Therefore, the loss of forced cooling has the same radiological consequences whether the Offsite PMF Berm remains in service or not.

(4) The postulated removal of the Offsite PMF Berm results in a small increase in the maximum water (flood) levels at SONGS 2&3. This increase in water level does not impact any structures, systems, and components that might increase the consequences of a failure of the pump or loss of power to the pump. Therefore, the loss of forced cooling has the same radiological consequences whether the Offsite PMF Berm remains in service or not.

(5) The proposed activity will not create the possibility for an accident of a different type than any previously evaluated in the UFSAR because the failure of the SFP cooling system (due to component failure, loss of offsite electrical power, etc.) would result in an accident or event (Spent Fuel Pool Boiling Accident) previously evaluated in the UFSAR.

(6) The removal of the Offsite PMF Berm and Channel from service does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because the small increase in the maximum water (flood) levels at SONGS 2&3 due to removal of the Offsite PMF Berm and Channel from service does not create the possibility of a malfunction with a result different than what was previously evaluated in the UFSAR.

(7) Removal of the Offsite PMF Berm and Channel from service will not affect the fuel rod fission product barrier because there is no impact to irradiated fuel stored in the SFP that might be caused by a small increase in water levels during a PMF event.

(8) Removal of the Offsite PMF Berm and Channel from Service using a revised analysis does not result in a departure from the methods of evaluation used to establish the design basis because computer software that was used to evaluate the flood effects to SONG 2&3 due to postulated removal of the Offsite PMF Berm is an acceptable NRC method for assessing flooding hazards at a nuclear power plant in accordance with NEI 96-07 Section 3.4.

10CFR50.59 Evaluation 202526642-15, Delete Licensee Controlled Specification's 3.3.112 and 3.7.118

Description:
The proposed activity deletes Licensee Controlled Specification's 3.3.112, "Fuel Handling Isolation Signal (FHIS)" and 3.7.118 "Fuel Handling Building Post-Accident Cleanup Filter System". This supports permanent removal from service of the Fuel Handling Isolation Signal, the Fuel Handling Building Post-Accident Cleanup Units(PACU) and associated equipment in accordance with the SONGS Decommissioning (DEC) process. Changes are also made to the appropriate Sections of the UFSAR to reflect removal of the FHIS and the PACU Units from the Design Bases.

Evaluation Summary:

(1) Review of credible defueled condition events showed that the Fuel Handling Isolation Signal and the Post-Accident Cleanup Units do not directly or indirectly initiate any analyzed accidents. Therefore, the Removal from Service of the Fuel Handling Isolation Signal (FHIS) and the Fuel Handling Building Post-Accident Cleanup Units (PACUs) does not introduce the possibility of a change in the frequency of an accident because the FHIS and the PACUs are not an initiator of any accidents and no new failure modes are introduced.

(2) The malfunction of FHIS or a PACU Unit is not a previously evaluated condition as an initiator of any accidents. In addition, the Removal from Service of the PACU Units and FHIS would not increase the likelihood of occurrence of the malfunction of other SSCs that would initiate an accident.

(3) The dose consequence of an accident in the Fuel Handling Building previously evaluated would not be increased by the proposed activity because in all applicable accident analysis, FHIS and the PACU Units are not credited with preventing or mitigating the consequences of accidents that could result in potential offsite exposure. This includes no credit for Fuel Handling Building Isolation Signal or filtration by the PACU Units.
(4) Removing FHIS and the PACU Units from service would not increase the consequences of an SSC malfunction because the UFSAR Accident Analyses assume that the FHIS and the PACU Units provide no function in reducing the radiological consequences of fuel handling accidents.

(5) The permanent removal from service of the FHIS and the PACU Units does not introduce the possibility of a new accident because the removal from service is not an initiator of any accident and no new failure modes are being introduced.

(6) The permanent removal from service of FHIS and the PACU Units does not introduce the possibility for a malfunction of an SSC with a different result because the change does not introduce a new failure mode.
(7) This change does not result in a Design Basis Limit for a Fission Barrier as described in the UFSAR being exceeded or altered because permanent removal from service of FHIS and the PACU Units is not associated with a fission product barrier.

(8) Removing FHIS and the PACU Units permanently from service will not result in a departure from the method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis. The offsite dose consequences calculated in the applicable accident analysis already assume that the PACU Units and FHIS do not function and provide no mitigating actions.

ENCLOSURE 2

SAN ONOFRE NUCLEAR GENERATION STATION

ISFSI

FACILITY CHANGE REPORT (FCR)

10 CRF 72.48 EVALUATION SUMMARIES

FOR THE PERIOD

FROM APRIL 1, 2015 THROUGH April 1, 2017

10CFR72.48 Evaluation 801372566-280, HOLTEC ISFSI Pad

Description:
The plant modifications, as described in detail in NECP 801372566, perform the work steps to excavate and then construct a new ISFSI utilizing the Holtec International HI-STORM UMAX Canister Storage System (HI-STORM UMAX) (referred to as ISFSI Pad 3) at SONGS. ISFSI Pad 3 will be located in the proximate area directly South and West of existing ISFSI Pads 1 and 2, the NUHOMS Horizontal Modular Storage System.

Evaluation Summary:

(1) The accident scenarios evaluated in DCS-001, SONGS Dry Spent Fuel Storage Project 10CFR72.212 Evaluation, and in SO1-207-1-M135, Updated Final Safety Analysis Report for the Standardized Advanced NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel, are all caused by natural phenomena or forces external to the cask system There is no increase in frequency of occurrence of any accident or hazard because frequency of accidents which involve the change (e.g. seismic event and drop accidents) are postulated to occur with a frequency of 1.0. The frequency of accidents which involve the change (e.g. seismic event and drop accidents) are already assumed to occur with a frequency of 1.0. Therefore, there is no increase in frequency of occurrence of any accident or hazard.

(2) The proposed activity does result in some cases where the NUHOMS storage module slides slightly further as quantified. However, the slight increase in sliding distance still maintains a large amount of margin from sliding off of the ISFSI pad and the pad structural capacity remains sufficient to support the module in all analyzed cases. Therefore, the proposed activity will not result in an increase in the likelihood of occurrence of a malfunction.

(3) The minimal increases in sliding of the NUHOMS Modules on the ISFSI does not increase in the consequences of an accident previously evaluated in the FSAR as the module remains on the ISFSI pad and in an ultimately safe and analyzed condition. Additionally, the structural capacity of the pad remains sufficient in all analyzed cases to support the system as described in the licensing basis calculations.

(4) The minimal increase in vertical impact, sliding and acceleration forces do not have any effect on the consequences of a malfunction of either of the SSCs. The forces are bounded by licensing basis analysis and the NUHOMS system remains on the ISFSI pad as required during a bounding seismic event.

(5) The modules are shown to keep within the bounds of the ISFSI pad, and there is no other effect of the excavation that could cause any other different behavior of the system. Therefore the proposed activity will not create a possibility for an accident of a different type.

(6) The proposed activity will not reduce the structural capacity of the ISFSI Pad or the response of the NUHOMS system beyond acceptable limits. Therefore, there is no possibility for the creation of a

malfunction with a different result.

(7) No design basis limit for a fission product barrier is altered or exceeded because the proposed activity does not decrease the structural integrity of the canister or increase component temperatures (including fuel cladding).

(8) The proposed activity does not involve a modification that affects a method of evaluation because there are no changes to the licensing bases analysis (i.e. drops and seismic accelerations) as the licensing basis vertical drop scenario.

ENCLOSURE 3

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

TECHNICAL SPECIFICTION BASES CHANGES

FOR THE PERIOD

FROM APRIL 1, 2015 THROUGH April 1, 2017

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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 and LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. The Completion Times of the Required Actions are also applicable when a specified Condition in the Applicability is entered intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise the safe storage of irradiated fuel. Intentional entry into ACTIONS should not be made for convenience.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed in order to verify the facility conditions are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.
SR 3.0.2	SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing Surveillance or maintenance activities). The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.
SR 3.0.3	SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

BASES

SR 3.0.3 (continued)

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on risk (from delaying the Surveillances) and impact on any analysis assumptions, in addition to facility conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10CFR50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

BASES

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary.

B 3.1 PLANT SYSTEMS

B 3.1.1 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2, Reference 1, and the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.3.4 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The resultant dose to a person at the exclusion area boundary or low population zone is a small fraction of the 10 CFR 50.67 (Ref. 5) limits. According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, however, there would be < 23 ft of water above the top of the bundle. The fuel storage pool water level satisfies Criterion 3 of the NRC Policy Statement.
LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of fuel assemblies (i.e., irradiated fuel, non-irradiated fuel, and the dummy fuel assembly) in the fuel storage pool since the potential for a release of fission products exists.

BASES (continued)

ACTIONS	A.1 When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.
REFERENCES	1. UFSAR, Section 9.1.2. 2. UFSAR, Section 9.1.3. 3. UFSAR, Section 15.7.3.4. 4. Regulatory Guide 1.183. 5. 10 CFR 50.67.

B 3.1 PLANT SYSTEMS

B 3.1.2 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	As described in LCO 3.1.3, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment, discharge burnup, and cooling time (plutonium decay). Although the water in the spent fuel pool is normally borated to ≥ 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron while maintaining $K_{eff} < 1.0$. Credit for boron is taken to maintain $K_{eff} \leq 0.95$.
APPLICABLE SAFETY ANALYSES	Soluble boron in the spent fuel pool is credited in criticality analyses for normal and accident conditions. The relevant accidents are 1) Fuel Assembly Dropped Horizontally On Top of the Racks, 2) Fuel Assembly Dropped Vertically Into a Storage Location Already Containing a Fuel Assembly, 3) Fuel Assembly Dropped to the SFP Floor, and 4) Fuel Misloading in either Region I or Region II. The limiting accident is Fuel Misloading in either Region I or Region II. A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.1.3 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the misloading of one fresh assembly with the maximum permissible enrichment. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by the postulated accident scenario. Under normal, non-accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 970 ppm. Under accident conditions, the soluble boron needed to maintain K_{eff} less than or equal to 0.95, including uncertainties, is 1700 ppm. A SFP boron dilution analysis shows that dilution from 2000 ppm to below 1700 ppm is not credible. Therefore, the minimum required soluble boron concentration is 2000 ppm. The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of the NRC Policy Statement.
LCO	The specified concentration of 2000 ppm dissolved boron in the fuel pool preserves the assumptions used in the analyses described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

BASES (continued)

ACTIONS	A.1 and A.2 When the concentration of boron in the spent fuel pool is less than required 2000 ppm, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to the required 2000 ppm.
SURVEILLANCE REQUIREMENTS	SR 3.1.2.1 This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.
REFERENCES	1. UFSAR, Section 9.1.

B 3.1 PLANT SYSTEMS

B 3.1.3 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 1542 fuel assemblies. Two types/sizes of spent fuel storage racks are used (Region I and Region II). The two Region I racks each contain 156 storage locations each spaced 10.40 inches on center in a 12x13 array. Four Region II storage racks each contain 210 storage locations in a 14x15 array. The remaining two Region II racks each contain 195 locations in a 13x15 array. All Region II locations are spaced 8.85 inches on center.

To maintain $K_{eff} < 0.95$ for spent fuel of maximum enrichment up to 4.8 w/o, (1) soluble boron is credited, and (2) the following storage patterns and borated stainless steel guide tube inserts are used as needed:

(1) unrestricted storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(2) SFP Peripheral storage, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(3) 2x2 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(4) 3x3 storage patterns, minimum discharge burnup and cooling time requirements vs. initial enrichment,

(5) credit for inserted Control Element Assemblies (CEAs),

(6) credit for erbia in fresh assemblies,

(7) credit for cooling time (Pu-241 decay), and,

(8) credit for borated stainless steel guide tube inserts.

When soluble boron is credited, the following acceptance criteria apply:

(1) Under normal conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than 1.0 when flooded with unborated water, and,

BASES (continued)

BACKGROUND (continued)

(2) Under normal and accident conditions, the 95/95 neutron multiplication factor (K_{eff}), including all uncertainties, shall be less than or equal to 0.95 when flooded with borated water.

APPLICABLE SAFETY ANALYSES

The spent fuel storage facility is designed for noncriticality by use of adequate spacing, neutron absorbing stainless steel cans, borated water with a minimum soluble boron concentration of 970 ppm, and storage of fuel assemblies in accordance with the administrative controls in LCO 3.1.3 and LCS 4.0.100, "Fuel Storage Patterns".

The spent fuel pool storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in the accompanying LCO, ensure that the K_{eff} of the spent fuel pool will always remain < 1.00 under normal, non-accident conditions assuming the pool to be flooded with unborated water. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under normal, non-accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 970 ppm. The K_{eff} of the spent fuel pool will always remain ≤ 0.95 under accident conditions assuming the pool to be flooded with borated water with a minimum soluble boron concentration of 1700 ppm. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Regions I and II of the spent fuel pool.

ACTIONS

A.1

When the configuration of fuel assemblies stored in Regions I and II of the spent fuel pool is not in accordance with LCO 3.1.3, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR verifies by administrative means that the fuel assembly is stored in accordance with LCO 3.1.3 or Design Features 4.3.1.1, or LCS 4.0.100. For fuel not stored in accordance with LCO 3.1.3, performance of this SR will ensure compliance with Specification 4.3.1.1.

This surveillance is performed prior to the initial storage of a fuel assembly in the spent fuel pool location and prior to each subsequent movement to a new location.

BASES (continued)

REFERENCES UFSAR, Section 9.1.2.2.