NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

October 23, 2006

NRC 2006-0040
10 CFR 50.90

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

License Amendment Request 246 - Deletion of Core Alterations

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC), hereby submits a proposed amendment to the Technical Specifications (TS) for Point Beach Nuclear Plant (PBNP), Units 1 and 2 to eliminate the use of the term CORE ALTERATIONS in the Technical Specifications. Suspending CORE ALTERATIONS has no effect on the initial conditions or mitigation of any design basis accident or transient. Therefore, the uses of the defined term CORE ALTERATIONS are proposed to be removed from the Technical Specifications.

The proposed amendment incorporates changes reflected in Technical Specification Task Force (TSTF) 471-T, Revision 1, "Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes". The proposed amendment also makes additional changes from TSTF-51-A, Revision 2, "Revise containment requirements during handling irradiated fuel and core alterations".

The proposed change to Technical Specification Bases Section B 3.8.10 represents an administrative change. The deletion of CORE ALTERATIONS from the corresponding Limiting Condition for Operation Section 3.8.10 was approved and issued under Amendments 201 and 206 dated August 8, 2001. At the time of the submittal (NPL-99-0669, dated November 15, 1999), the corresponding changes to the Technical Specification Bases Section B 3.8.10 were inadvertently omitted.

This submittal satisfies a corrective action contained in PBNP Licensee Event Report 50-266/2005-007, "Control Rod Movement with Refueling Cavity Water Below TS 3.9.6 Limit", dated January 16, 2006.

The applicable Technical Specification Bases are included in this submittal for completeness.

Enclosure I contains the description and analysis of the proposed change and the Significant Hazards Evaluation. Enclosure II contains the existing TS pages marked up to show the proposed change. Enclosure III is a markup of the applicable TS Bases.

NMC requests approval of the proposed license amendment by March 2007 in support of the upcoming Point Beach Unit 1 refueling outage scheduled to commence on April 1, 2007, with the amendment being implemented within 30 days.

Summary of Commitments

This letter contains no new commitments or revisions to existing commitments.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on October 23, 2006.

Dennis L. Koehl

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures: I - Description and Analysis of Changes
II - Proposed Technical Specification Changes
III - Proposed Technical Specification Bases

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE I

DESCRIPTION AND ANALYSIS OF CHANGES

LICENSE AMENDMENT REQUEST 246
DELETION OF CORE ALTERATIONS

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

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1. DESCRIPTION

This letter requests an amendment to License DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP), Units 1 and 2 Technical Specifications, to eliminate the use of the defined term CORE ALTERATIONS. Suspending core alterations has no effect on the initial conditions or mitigation of any design basis accident or transient. Therefore, the use of the defined term CORE ALTERATIONS is proposed to be removed from the Technical Specifications. The proposed amendment incorporates changes reflected in Technical Specification Task Force (TSTF) 471-T, Revision 1.

The proposed amendment makes additional changes from TSTF-51-A. TSTF-51-A eliminated all uses of the defined term CORE ALTERATIONS from Applicability statements and most uses of the term in Required Actions. TSTF-471-T eliminates the remaining few instances of the defined term CORE ALTERATIONS. PBNP proposes adopting the necessary portions of TSTF-51-A and TSTF-471-T to eliminate the defined term CORE ALTERATIONS from the Technical Specifications.

2. PROPOSED CHANGES

The proposed license amendment would revise the PBNP Technical Specifications to eliminate the use of the defined term CORE ALTERATIONS. The proposed changes, which are indicated on the marked up pages in Enclosures II and III, are described below:

a. Technical Specification 1.1, Delete the definition of CORE ALTERATION.
b. Technical Specification 3.3.5, Control Room Emergency Filtration System (CREFS) Actuation Instrumentation (CREFS), Required Action B.1 is deleted, resulting in the renumbering of subsequent Required Actions.
   Table 3.3.5-1, delete Note b.
c. Technical Specification 3.7.9, Control Room Emergency Filtration System (CREFS), delete During CORE ALTERATIONS from Applicability. Delete Required Action B.1, Suspend CORE ALTERATIONS, resulting in the renumbering of the subsequent Required Actions.
d. Technical Specification 3.9.1, Boron Concentration, Required Action A.1 is deleted, resulting in the renumbering of the subsequent Required Actions.
e. Technical Specification 3.9.2, Nuclear Instrumentation, Required Action A.1 is deleted, resulting in the renumbering of the subsequent Required Actions.

f. Technical Specification 3.9.6, Refueling Cavity Water Level, the phrase "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts," is deleted from the Applicability Statement. Also, Required Action A.1 is deleted, resulting in the renumbering of Required Action A.2.

g. Bases B 3.3.5, Revised in accordance with the changes proposed in the applicable TS 3.3.5-1.

h. Bases B 3.7.9, Revised in accordance with the changes proposed in the applicable TS 3.7.9.

i. Bases B 3.8.10, Revised in accordance with the changes approved as part of the Improved Technical Specifications (ITS) amendment. (Amendments 201/206, dated August 8, 2001).

j. Bases B 3.9.1, Revised in accordance with the changes proposed in the applicable TS 3.9.1.

k. Bases B 3.9.2, Revised in accordance with the changes proposed in the applicable TS 3.9.2.

l. Bases B 3.9.3, Revised in accordance with the changes proposed in the applicable TS 3.9.3.

m. Bases 3.9.6, Revised in accordance with the changes proposed in the applicable TS 3.9.6.

3. BACKGROUND

The term CORE ALTERATION is defined in the PBNP Technical Specifications as, "CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position."

The proposed change will eliminate the defined term CORE ALTERATION from the Technical Specifications. Suspending core alterations has no effect on the initial conditions or mitigation of any design basis accident or transient. Therefore, the uses of the defined term CORE ALTERATIONS are proposed to be removed from the Technical Specifications.

Currently, PBNP procedures include precautions to ensure that no control rod drive shafts are still latched to avoid accidentally lifting a control rod and potentially violating a Required Action which prohibits CORE ALTERATIONS. These actions have no safety benefit as the shutdown margin is determined assuming the worst configuration of control rods.

4. TECHNICAL ANALYSIS

The term "CORE ALTERATION" does not appear in the Standard Review Plan or in the Title 10 Code of Federal Regulations. Since CORE ALTERATIONS only occur when the reactor vessel head is removed, it only applies in MODE 6. There are only two accidents considered during MODE 6 (with the reactor vessel head removed) for pressurized water reactors: a fuel handling accident and a boron dilution accident. According to the Standard Review Plan, a fuel handling accident is initiated by the dropping of an irradiated fuel assembly, either in the Containment or in the Fuel Building. There are no credited mitigation actions, except for taking credit for control room ventilation system to reduce the dose consequences. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of a fuel handling accident.

The second analyzed event is a boron dilution accident. A boron dilution accident is initiated by a dilution source which results in the boron concentration dropping below that required to maintain the SHUTDOWN MARGIN. As described in the Bases of Specification 3.9.1, "Boron Concentration," (which applies in MODE 6), the refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of Keff ≤ 0.95 during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures. The accident is mitigated by stopping the dilution. Suspension of CORE ALTERATIONS has no effect on the mitigation of a boron dilution accident. Movement of control rods or fuel do not affect the initial conditions of a boron dilution accident as it is assumed that the control rods and fuel are in the most adverse conditions with a large safety margin (Keff ≤ 0.95). To address the possibility of a misloaded fuel assembly in Technical Specification 3.9.2, a Required Action exists that suspends positive reactivity additions if nuclear instrumentation is not available. This precludes movement of fuel assemblies which could add reactivity to the core.

As described in TSTF-51, since the only accident postulated to occur during CORE ALTERATIONS that results in significant radioactive release is the fuel handling accident (FHA), the proposed Technical Specification requirements omitting CORE ALTERATIONS is justified. For PBNP, the FHA analysis submitted as part of the selective application of Alternative Source Term (AST) in NMC to NRC letter (NRC 2003-0028) dated March 27, 2003, also describes that during CORE ALTERATIONS, only a postulated fuel handling accident results in cladding damage and a potential radiological release.

In summary, with the exception of suspending movement of irradiated fuel assemblies, there are no design basis accidents (DBAs) or transients that are initiated by, or mitigation affected by, suspension of CORE ALTERATIONS. Therefore, if all Required Actions that require suspension of CORE ALTERATIONS also require suspension of movement of irradiated fuel, suspension of CORE ALTERATIONS provides no additional safety benefit.

5. NO SIGNIFICANT HAZARDS CONSIDERATION

NMC is proposing an amendment to the PBNP Technical Specifications that will eliminate the use of the term CORE ALTERATIONS. The proposed changes have been evaluated against the standards in 10 CFR 50.92 and have been determined to not involve a significant hazards consideration in that:

5.1 Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change eliminates the use of the defined term CORE ALTERATIONS from the Technical Specifications. CORE ALTERATIONS are not an initiator of any accident previously evaluated except a fuel handling accident. The revised Technical Specifications that protect the initial conditions of a fuel handling accident also require the suspension of movement of irradiated fuel assemblies, which protects the initial condition of a fuel handling accident.

Therefore, suspension of CORE ALTERATIONS do not affect the initiators of the accidents previously evaluated and suspension of CORE ALTERATIONS does not affect the mitigation of the accidents previously evaluated.

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

5.2 Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical modification of the plant (i.e., no new or different type of equipment will be installed) or

a significant change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions.

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

5.3 Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Only two accidents are postulated to occur during plant conditions where CORE ALTERATIONS may be made: a fuel handling accident and a boron dilution accident. Suspending movement of irradiated fuel assemblies prevents a fuel handling accident. Also, requiring the suspension of CORE ALTERATIONS is redundant to suspending movement of irradiated fuel assemblies and does not increase the margin of safety. CORE ALTERATIONS have no effect on a boron dilution accident. Core components are not involved in the initiation or mitigation of a boron dilution accident. Therefore, CORE ALTERATIONS have no effect on the margin of safety related to a boron dilution accident.

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

Based on the above evaluation, NMC has concluded that the proposed amendment involves no significant hazards considerations under the standards set forth in 10 CFR 50.92(c). Accordingly, a finding of "no significant hazards consideration" is justified.

6. ENVIRONMENTAL CONSIDERATION

NMC has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, NMC concludes that the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

7. PRECEDENTS

Calvert Cliffs submitted the lead plant application for TSTF-471-T (Adams Accession Number ML051660207, dated June 7, 2005). In preparing this request, NMC incorporated TSTF-471-T, Rev. 1 as submitted for Calvert Cliffs. In addition, the Calvert Cliffs NRC Request for Additional Information dated March 30, 2006, and the corresponding response dated May 12, 2006, were evaluated and applicable information was included in this submittal. The NRC has approved the Calvert Cliffs lead plant submittal for TSTF-471 on September 21, 2006 (Federal Register Notice 71FR59533).

ENCLOSURE II

PROPOSED TECHNICAL SPECIFICATION CHANGES
(Marked-up Pages)

LICENSE AMENDMENT REQUEST 246
DELETION OF CORE ALTERATIONS

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
~~CORE ALTERATION~~	~~CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.4. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988.
E - AVERAGE DISINTEGRATION ENERGY	E shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

3.3 INSTRUMENTATION

3.3.5 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

LCO 3.3.5 The CREFS actuation instrumentation for each Function in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions inoperable.	A.1 Place CREFS in the emergency mode of operation.	7 days
B. Required Action and associated Completion Time not met.	----------------NOTE---------------- Required Actions~~s~~ B.1 ~~and B.2 are~~ **is** not applicable for inoperability of the Containment Isolation actuation function. --	
	~~B.1~~ ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	~~B.2~~ Suspend movement of irradiated fuel assemblies.	Immediately
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.~~3~~ **2**	Be in MODE 3.	6 hours
	AND		
	B.~~4~~ **3**	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.5-1 to determine which SRs apply for each CREFS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.2	Perform COT.	92 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.5-1 (page 1 of 1)
CREFS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Control Room Radiation				
	a. Control Room Area Monitor	1, 2, 3, 4, (a)~~, (b)~~	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3	NA
	b. Control Room Air Intake	1, 2, 3, 4, (a)~~, (b)~~	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.3	NA
2.	Containment Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

~~(b) During CORE ALTERATIONS~~.

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Filtration System (CREFS)

LCO 3.7.9 CREFS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4,
During movement of irradiated fuel assemblies~~,~~
~~During CORE ALTERATIONS~~.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. CREFS inoperable.	A.1	Restore CREFS to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	~~B.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	~~B.2~~	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	B.~~3~~2	Be in MODE 3.	6 hours
	AND		
	B.~~4~~3	Be in MODE 5.	36 hours

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	~~A.2~~	Suspend positive reactivity additions.	Immediately
	AND		
	A.~~3~~**2**	Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

AND

One source range audible count rate circuit shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	~~A.1~~ A.1	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	~~A.2~~	Suspend positive reactivity additions.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	Once per 12 hours
C. Required source range audible count rate circuit inoperable.	C.1	Initiate action to isolate unborated water sources.	Immediately

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	~~A.2~~	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

ENCLOSURE III

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(Marked-up Pages)

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DELETION OF CORE ALTERATIONS

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

B 3.3 INSTRUMENTATION

B 3.3.5 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

BASES

BACKGROUND	The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. The control room ventilation system normally operates in the normal operating mode (Mode 1). Upon receipt of an actuation signal, the CREFS initiates the emergency make-up (Mode 4) mode of operation. The control room ventilation system and its operating modes are described in the Bases for LCO 3.7.9, "Control Room Emergency Filtration System." The actuation instrumentation consists of noble gas radiation monitor in the air intake and control room area radiation monitor. A high radiation signal from either of these detectors will initiate the emergency make-up mode of operation (Mode 4) of the CREFS.
APPLICABLE SAFETY ANALYSES	The CREFS provides airborne radiological protection for control room personnel, as demonstrated by the limiting control room dose analyses for the design basis large break loss of coolant accident. Control room dose analysis assumptions are presented in the FSAR, Section 14.3.5 (Ref. 1). In MODES 1, 2, 3, and 4, the CREFS radiation monitor actuation signal will provide automatic initiation of CREFS in the emergency make-up mode of operation (Mode 4) during design basis events which result in significant radiological releases to the environs (e.g. large break loss of coolant accident, steam generator tube rupture, reactor coolant pump locked rotor, etc;). The CREFS radiation monitor actuation signal also provides automatic initiation of CREFS, in the emergency make-up mode of operation (Mode 4), to assure control room habitability in the event of a fuel handling **accident** during movement of irradiated fuel. ~~and CORE ALTERATIONS~~ Further Applicable Safety Analysis information for CREFS is contained in the Bases for LCO 3.7.9, "Control Room Emergency Filtration System." The CREFS actuation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

BASES

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREFS is OPERABLE.

1. Control Room Radiation

The LCO requires the control room area (RE-101) and the control room air intake noble gas monitor (RE-235) to be OPERABLE, to ensure that the instrumentation necessary to initiate the CREFS emergency make-up mode (Mode 4) is OPERABLE.

Table 3.3.5-1 identifies the Technical Specification Trip Setpoint for the Control Room Area Monitor and Control Room Air Intakes as not applicable (NA). No Analytical Value is assumed in the accident analysis for these functions. The nominal setting required for the Control Room Area Monitor is 5 mr/hr and the nominal setting for the Control Room Air Intakes is 5E-5 μCi/cc. These nominal settings were developed outside of the setpoint methodology.

2. Containment Isolation

Refer to LCO 3.3.2, Function 3, for all initiating Functions and requirements.

APPLICABILITY

The CREFS Functions must be OPERABLE in MODES 1, 2, 3, 4, and during ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

The Applicability for the CREFS actuation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1 in the accompanying LCO. The Completion Time(s) of the inoperable Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the containment isolation signal, control room area radiation monitor (RE-101) and the control room intake noble gas monitor (RE-235).

BASES

ACTIONS (continued)

If a Function is inoperable, 7 days is permitted to restore the Function to OPERABLE status from the time the Condition was entered for that Function. The 7 day Completion Time is the same as for inoperable CREFS. The basis for this Completion Time is the same as provided in LCO 3.7.9. If the Function cannot be restored to OPERABLE status, CREFS must be placed in the emergency make-up mode of operation (MODE 4). Placing CREFS in the emergency make-up mode of operation accomplishes the actuation instrumentation's safety function.

B.1, B.2 **and**~~,~~ B.3~~, and B.4~~

Condition B applies when the Required Action and associated Completion Time for Condition A have not been met. If movement of irradiated fuel assemblies ~~or CORE ALTERATIONS are~~ **is** in progress, ~~these~~ **this** ~~activities~~ **activity** must be suspended immediately to reduce the risk of accidents that would require CREFS actuation. In addition, if any unit is in MODE 1, 2, 3, or 4, the unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions for Condition B are modified by a Note that states that Required Action~~s~~ B.1 ~~and B.2 are~~ **is** not applicable for inoperability of the Containment Isolation actuation function. This note is necessary because the Applicability for the Containment Isolation actuation function is Modes 1, 2, 3, and 4. The Containment Isolation actuation function is not used for mitigation of accidents involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which CREFS Actuation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, in the case of the control room area and control room intake noble gas monitors, no independent instrument channel exist, therefore, the CHANNEL CHECK for these monitors will consist of a qualitative assessment of expected channel behavior, based on current plant and

BASES

LCO (continued)

e. Ductwork and dampers are OPERABLE, and air circulation can be maintained; and

f. CREFS is capable of being manually initiated in the emergency make-up mode of operation (mode 4).

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS~~, CREFS must be OPERABLE to control operator exposure during and following a DBA.

During movement of irradiated fuel assemblies ~~and CORE ALTERATIONS~~, the CREFS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

When CREFS is inoperable, action must be taken to restore the system to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of a DBA challenging control room habitability occurring during this time period.

B.1, B.2, **and** B.3~~, and B.4~~

If CREFS cannot be restored to OPERABLE status within the required Completion Time with ~~CORE ALTERATIONS or~~ movement of irradiated fuel in progress, ~~these activities~~ **this activity** must be suspended immediately. Immediately suspending ~~these activities~~ **this activity** places the unit in a condition that minimizes risk from ~~these activities~~ **this activity**. This does not preclude the movement of fuel to a safe position.

In MODE 1, 2, 3, or 4, if CREFS cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

LCO (continued)	This configuration is considered acceptable for a limited period of time based on maintaining all required redundant shared equipment and their associated power sources for the unit in MODE 1, 2, 3, or 4 in an OPERABLE status, retaining redundancy in residual heat removal for the unit in MODE 5 or 6, in addition to the probability for an event resulting in a bus fault or loss of offsite power with a failure of the bus cross tie breaker to open. If any tie breakers is closed outside of the allowances outlined above, the affected electrical power distribution buses are inoperable. Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown.
APPLICABILITY	The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6 provide assurance that: a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core; b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition. The AC, DC, and AC vital instrument bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.
ACTIONS	<u>A.1, A.2</u> Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features.~~ to allow continuation of CORE ALTERATIONS and fuel movement.~~. Declaring the required features associated with an inoperable distribution subsystem inoperable ensures that the appropriate restrictions are implemented in accordance with the affected supported features LCO Required Actions.

BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

ACTIONS

A.1 ~~and A.2~~

Continuation of ~~CORE ALTERATIONS or~~ positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron

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BASES

ACTIONS (continued)

concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving ~~CORE ALTERATIONS or~~ positive reactivity additions must be suspended immediately.

Suspension of ~~CORE ALTERATIONS and~~ positive reactivity additions shall not preclude moving a component to a safe position.

A.~~3~~2

In addition to immediately suspending ~~CORE ALTERATIONS or~~ positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration is determined periodically by chemical analysis of a representative sample of the interconnected volumes.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. FSAR. Sections 1.3.5, 3.1, and 9.3.
2. FSAR. Chapter 14.1.4.

BASES

LCO (continued)	To be OPERABLE, each monitor must provide visual indication in the control room. In addition, at least one of the two monitors must provide an OPERABLE audible count rate function in the control room to alert operators to the initiation of a boron dilution event.
APPLICABILITY	In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, the installed BF3 source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 ~~and A.2~~

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, ~~CORE ALTERATIONS and~~ positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity.

However, since ~~CORE ALTERATIONS and~~ positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

BASES

LCO (continued)

Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure specified in the FSAR can be achieved.

The containment personnel airlock doors may be open during movement of recently irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls to implement closure will be in effect when maintaining airlock doors open. Hoses and cables running through an airlock shall employ a means to allow safe, quick disconnection or severance. No fuel movement may occur prior to the minimum decay time of 65 hours. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment.

The allowance to have containment personnel airlocks open during fuel movement of recently irradiated fuel assemblies is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of recently irradiated fuel assemblies within containment is not being conducted, the potential for a limiting fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 ~~and A.2~~

If the containment equipment hatch, air locks, or any containment Purge and Exhaust System penetration is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of ~~these~~ **this** action~~s~~ shall not preclude completion of movement of a component to a safe position.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies ~~or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to 10 CFR 50.67 limits, as provided by the guidance of Reference 1.
APPLICABLE SAFETY ANALYSES	During ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment (Ref. 1). A minimum water level of 23 ft allows an overall decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.85% of the iodine released from the damaged fuel assembly gaps is elemental and the remainder is organic. Therefore, an overall decontamination factor of 200 is achieved if an elemental decontamination factor of 285 is assumed (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 65 hours prior to fuel handling without the containment penetration requirements of LCO 3.9.3, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 3). Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

BASES

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 1.

APPLICABILITY

LCO 3.9.6 is applicable ~~during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and~~ when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. LCO 3.9.3 provides additional requirements for movement of irradiated fuel assemblies within containment whose decay time is less then 65 hours. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.10, "Fuel Storage Pool Water Level."

ACTIONS

A.1 ~~and A.2~~

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of ~~CORE ALTERATIONS and~~ fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.