PACKAGE 3.8

ELECTRICAL POWER SYSTEMS

PART F

JUSTIFICATION FOR DIFFERENCES
(JFD)

from

NUREG-1431
IMPROVED STANDARD TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART F

PACKAGE 3.8

ELECTRIC POWER SYSTEMS

JUSTIFICATION FOR DIFFERENCES FROM IMPROVED STANDARD TECHNICAL SPECIFICATIONS (NUREG-1431) AND BASES

See Part E for specific proposed wording and location of referenced deviations.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	100	During the development certain wording preferences, English conventions, reformatting, renumbering, providing additional descriptive information as related to Prairie Island (PI), or editorial rewording consistent with plant specific nomenclature, system names, design, or current licensing basis were adopted. Bases for LCO 3.8.1, 3.8.2, 3.8.3, 3.8.4, 3.8.5, 3.8.6, 3.8.7, 3.8.8, 3.8.9, and 3.8.10 have been revised to add specific details from the CTS, design manuals, and P&IDs, and Surveillance details. As a result, the Technical Specifications (TS) should be more readily readable by, and therefore understandable to, plant operators and other users. During this process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	102	NUREG-1431 contained optional wording or values in brackets. The optional information was not applicable to PI and therefore not retained in the ITS. As an example, ISTS LCO 3.8..7 includes a NOTE regarding disconnecting inverters to facilitate performance of a battery equalizing charge. This NOTE was not included since this is not done at PI.
PA	103	NUREG-1431 contained optional wording or values in brackets. The correct information, based on PIs Current Licensing Bases (CLB), was incorporated and brackets removed.
CL	104	NUREG-1431 LCO 3.8.1, Required Actions A.2 and associated Bases have been deleted. Required Action A.2 involved restrictions on loss of offsite power to one division and loss of features on the other division. PI design has two paths feeding each bus; if one path is inoperable, there is still a second independent path feeding the same bus. Therefore, the loss of one path does not render the train or bus inoperable. Two paths on the same train or bus would have to be inoperable to make the train or bus inoperable. These limitations are not imposed in the CTS, and are therefore not included in the conversion to the ITS.

Difference Category	Difference Number	Justification for Differences
CL	3.8-105	NUREG-1431 LCO 3.8.1, Required Action A.3 (PI A.2) and associated Completion Time, states , "Restore ... 72 hours AND 6 days from discovery of failure to meet LCO". CTS 3.7.B.2 states, " One of the two required paths ... may be inoperable for 7 days ". Therefore, based on PI CLB, the ISTS Completion Time has been revised to state, "Restore ... 7 days". The 7 days takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. The second Completion Time in ISTS Required Action A.3 states, "6 days from discovery of failure to meet LCO" is being deleted and discussed in JFD CL3.8-106. Maintaining CLB was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement. The associated Bases have also been revised to be consistent with the changes noted above.

Difference Category	Difference Number	Justification for Differences
CL	3.8-106	The Completion Time limit of "6 days from the discovery of failure to meet the LCO" was not included in the ITS since PI does not have this requirement in the CTS. The intent of adding this limit to the Completion Time is to prevent a plant from continuously being in the LCO without ever meeting the full LCO requirements. This abuse of the LCO can be adequately addressed in plant procedures. For the past three years, the Offsite qualified paths have been considered out of service in accordance with the ITS less than 1% of the time. Since the ITS only requires two qualified paths to be OPERABLE at any one time, plant design would ensure that this LCO would be maintained. From this data, it is evident that PI has not abused the use of Allowed Outage Times for the Offsite qualified paths. Including this statement in the ITS would only add confusion for the operators. In addition, the associated Bases have been deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	107	NUREG-1431 LCO 3.8.1, Required Action B.3 and associated Bases have been revised by adding a Note stating, "Completion of ACTIONS B.3.1 and B.3.2 are not required if DG inoperability is due to preplanned preventative maintenance or testing." This Note was added based on PI CLB as stated in CTS 3.7.B.1 footnote. ITS Condition B provides two different options (Required Action B.3.1 or B.3.2) when one DG is inoperable. Required Action B.3.1 requires a determination that the OPERABLE DG is not made inoperable due to a common cause failure and B.3.2 is the performance of SR 3.8.1.2 requiring the OPERABLE DG to be started and operated at specific conditions. In the event a DG is made inoperable for preplanned preventative maintenance, there would not be any common cause associated with the maintenance. If Required Action B.3.2 is taken, PI would consider the other DG to be OPERABLE provided it passed its last SR, it is not inoperable for other known reasons, and its SR is current. Requiring the performance of Required Action B.3.2, while the other DG is inoperable for preventative maintenance, would invoke additional and unnecessary starting and over testing of the other DG. In addition, PI would consider the other DG to be inoperable while it is being tested in accordance with SR 3.8.1.2. This would intentionally and unnecessarily place the plant in Condition E with a Completion Time of 2 hours and possible plant shutdown. This could cause initiation of unnecessarily cycling the plant or systems. Neither the NRC, manufacturer, nor the industry believe that

Difference Category	Difference Number	Justification for Differences
	3.8-	
CL	107	(continued) over testing the DGs is good practice nor does the additional testing provide any added assurance that the DG will perform its intended safety function when required. Upon completing the preventative maintenance, the subject DG must be demonstrated OPERABLE prior to returning it to service.
CL	108	NUREG-1431 LCO 3.8.1, Required Action B.4 and associated Completion Time, states , "Restore ... 72 hours AND 6 days from discovery of failure to meet LCO." CTS 3.7.B.1 states, "One diesel generator may be inoperable for 7 days ... . " Therefore, based on PI CLB, the ISTS Completion Time has been revised to state, "Restore ... 7 days." The 7 days takes into account the capacity and capability of the remaining diesel generator and AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. The second Completion Time states, "6 days from discovery of failure to meet LCO", has been deleted. The deletion of this Completion Time is discussed in JFD CL3.8-109. Maintaining CLB was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement. In addition, the associated Bases have been revised to be consistent with the above changes.

Difference Category	Difference Number 3.8-	Justification for Differences
	109	Not used.
CL	110	NUREG-1431 LCO 3.8.1.c states, "[Automatic load sequencer for Train A and Train B]". This statement was deleted since the load sequencer has been relocated to ITS LCO 3.3.4, and therefore not retained in this Specification. In addition, Bases 3.8.1 and 3.8.2, LCO statement has been changed by deleting bracketed information related to the load sequencers.
PA	111	NUREG-1431 LCO 3.8.1, Required Action D, Note states, " ... when Condition D is ... ." This Note was changed by replacing the word "when" with the word "if". Since the paths go to both trains, loss of a path will not de-energize either train, therefore not necessarily requiring entry into Condition D. This change was made for consistency with changes made in other Sections of the ITS (e.g., Section 3.7) and is based on plant design. In addition, the word "any" is being replaced with the word "either." PI only has two trains; therefore, "either" is more representative of the plant design. Similar word changes are made in LCO 3.8.2, Required Action A, Note. In addition, the Bases have been revised to be consistent with the above changes.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	112	NUREG-1431 LCO 3.8.1, Condition F and associated Bases have been deleted. This Condition is for when a load sequencer is inoperable. The load sequencer Actions and Surveillance have been relocated to ITS Section 3.3.4.
PA	113	NUREG-1431 LCO 3.8.1, Condition H (PI Condition G) states, "Three or more ... inoperable." has been revised to state, "Two DGs inoperable and one or more paths inoperable" OR "One DG inoperable and two paths inoperable." This change was made to more accurately reflect the PI design and be more specific as to the combination of inoperabilities between the DG and required offsite circuits. These scenarios reflect where entering LCO 3.0.3 would be appropriate.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	114	NUREG-1431 Bases 3.8.4, Background Section has been revised by deleting the statement, "from the design minimum charge ...", and Bases SR 3.8.4.6 (PI SR 3.8.4.2) has also been revised by deleting the statement, "from the design minimum charge state ... ." The design minimum charge is in reference to the charging capacity of the battery charger. The deletion of this statement is consistent with PI design bases. PI has purchased batteries with a design and capacity in support of the PI accident analysis.
PA	115	NUREG-1431 SR 3.8.1.2 and associated Bases have been revised by deleting the wording, "from standby conditions". At PI the DGs are started in accordance with manufacturer's recommendations requiring prelube, warm-up, loading, and shutdown. "Standby" conditions are not necessarily the same as manufacturer's conditions and could cause confusion that may result in DG damage or voiding of the warranties. Therefore, this SR is revised to align with the current way PI starts the DGs. This is consistent with PI CLB and CTS 4.6.A.1.e.

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CL	116	NUREG-1431 SR 3.8.1.2, Note 3 and 3.8.1.3, Note 1 have been revised by replacing the statement, "as recommended by the manufacturer" with "in consideration of the manufacturer's recommendations." This change is consistent with the PI CTS (4.6.A.1.e). In addition, SR 3.8.1.3 Bases has been revised replacing the phrase, "... equivalent of the maximum expected accident loads" with "manufacturer's recommended loads." This change is consistent with PI CLB and testing practices, which are based on test limitations provided by one of the manufacturers.
TA	117	Incorporated TSTF-286, Rev. 2.
CL	118	NUREG-1431 SR 3.8.1.4 requires that each day tank contains a specific amount of fuel oil. PI CTS 4.6.A.1.a only requires the fuel oil level to be verified and does not specify any specific amount or quantity for the day tanks. Therefore, stating a specific amount or quantity of fuel oil to be maintained is being deleted. The PI design uses level switches which sense low and high levels of fuel oil in the day tanks. The level switches are permanently set in order to supply enough fuel oil to the DGs in support of the USAR analysis. In addition, the associated Bases have been revised to be consistent with the above changes.

Difference Category	Difference Number 3.8-	Justification for Differences
	119	Not used.
TA	120	Incorporated TSTF-163, Rev. 2.
CL	121	NUREG-1431 SR 3.8.1.9 and associated Bases are being deleted since the load reject test requirements, per the CTS, are included in ISTS SR 3.8.1.10 (PI SR 3.8.1.7). Therefore, requiring them to be performed again is not necessary nor does it serve any advantage in evaluating system OPERABILITY.
CL	122	NUREG-1431 SRs 3.8.1.10 (PI 3.8.1.7), 3.8.1.13 (PI 3.8.1.8), 3.8.1.14 (PI 3.8.1.9), 3.8.4.6 (PI 3.8.4.2) Notes and associated Bases, have been revised by deleting the statement, "This Surveillance shall not be performed in MODE 1 or 2." PI CTS does not provide a restriction or MODE of Applicability for this SR. PI presently performs some of these SRs during the specified MODES and maintains to keep this flexibility in accordance with CLB.

Difference Category	Difference Number 3.8-	Justification for Differences
TA	123	NUREG-1431 SRs 3.8.1.10 (PI 3.8.1.7), 3.8.1.13 (PI 3.8.1.8), 3.8.1.14 (PI 3.8.1.9), 3.8.1.19 (PI 3.8.1.10), SR 3.8.4.6 (PI 3.8.4.2) and SR 3.8.4.7(PI 3.8.4.3). Notes and associated Bases, have been revised by deleting the statement, "However, credit may be taken for unplanned events that satisfy this SR." Deleting this sentence is consistent with approved TSTF-8, Rev. 1.
	124	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	125	NUREG-1431 SRs 3.8.1.10 (PI 3.8.1.7), 3.8.1.14 (PI 3.8.1.9) and associated Bases require that the DG be operating at its rated power factor when the total load rejection test is performed, and for 2 hours of the 24 hour full load test, respectively. The practice of performing these tests at the rated power factor has been determined to be unjustified, potentially destructive testing due to exceeding the vendors' recommendation for maximum voltage of the generator. The NRC has concurred with discontinuing the DG load rejection testing at rated power factor. Therefore, this requirement has been deleted. Also the Bases have been revised deleting the requirement of the "full load test" to be the "largest load test." PI does not perform a full load test as required by the ISTS. Instead, PI performs a test to ensure it can handle the largest load that will be required. These changes are consistent with PI USAR and CLB.
X	126	NUREG-1431 SRs and associated Bases contain Frequencies of 18 months. The SR Frequencies have been increased to 24 months to be consistent with PI refueling intervals.
	127	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	128	NUREG-1431 SR 3.8.1.11 and associated Bases have been deleted. PI does not perform this specific SR for verifying specific conditions resulting from an actual or simulated loss of offsite power signal. Instead, PI performs SR 3.8.1.19 using a simulated or actual loss of offsite power in conjunction with an actual or simulated SI actuation signal. Both SRs verify de-energization of emergency buses, load shedding from the emergency buses and specific DG performance from an auto start. The SR PI performs encompasses the testing being deleted under SR 3.8.1.11.
CL	129	NUREG-1431 SR 3.8.1.14, (PI SR 3.8.1.9.c) has been revised by adding specific CTS requirements (CTS SR 4.6.A.3.c), ensuring the DG achieves a specific steady state voltage and frequency. In addition, the associated Bases have been revised to be consistent with the above change.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	130	NUREG-1431 SR 3.8.1.15 and associated Bases are being deleted. Note 1 requires that the SR shall be performed within 5 minutes of shutting down the DG after the DG has operated under specific conditions. The PI CTS does not contain this requirement to restart the DG. The remainder of this SR (start time, voltage, and frequency) testing the DG, is performed under ISTS SR 3.8.1.7 (PI SR 3.8.1.6) at a Frequency of 184 days. Since ISTS SR 3.8.1.7 verifies the same parameters for DG OPERABILITY as does ISTS SR 3.8.1.15, PI does not see the need to perform a restart of the DG to again verify the same parameters. This would be excessive testing on the DG without providing any additional benefits. PI is deleting this SR based on CLB. Maintaining CLB requirements was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement.
PA	131	NUREG-1431 SR 3.8.1.16 and associated Bases have been deleted. PI CLB does not require synchronizing the DG with offsite power sources while loaded with emergency loads, nor transferring loads to offsite power sources. Deleting this SR is consistent with PI current operating and testing practices.

Difference Category	Difference Number 3.8-	Justification for Differences
	132	Not used.
CL	133	NUREG-1431 SR 3.8.1.20 and associated Bases have been deleted. This SR was to be performed on a 10 year interval. This SR is the same as ISTS SR 3.8.1.7 (PI SR 3.8.1.6), which is performed on a 184 day Frequency. Since the same information is being verified on a 184 day basis, there is no real benefit for maintaining a requirement or Frequency of 10 years.
PA	134	NUREG-1431 LCO 3.8.3, Actions Note and associated Bases have been revised by deleting "Separate Condition entry is allowed for each DG" statement. This is not needed since the ITS Condition provides adequate guidance and Required Actions when one or both DGs are inoperable.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	135	NUREG-1431 Bases 3.8.1, Background Section, has been revised by providing additional detail describing that the transformers are capable of block loading without load sequencing or load rejected. This additional detail more specifically describes the PI plant design and operation. Additional information was provided to more accurately describe the PI design for the DG, which buses they are dedicated to, and each unit's continuous service rating values.
CL	136	NUREG-1431 Bases 3.8.1, LCO Section has been revised by deleting the following, " Additional DG capabilities must be demonstrated to meet required Surveillance, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode." PI does not have any CLB requirement to ensure that the DGs have or maintain this capability. Therefore this statement can be deleted.
TA	137	TSTF 37, Rev. 2 was incorporated.
CL	138	NUREG-1431 SR 3.8.1.18 and associated Bases have been deleted. PI CLB does not require this surveillance testing and it is therefore being deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	139	NUREG-1431 SR 3.8.1.19 has been revised by deleting ISTS SR requirements c.2, c.3, c.4, and c.5. Requirement c.1 was edited to represent the PI CTS requirement. These additional requirements are not in the PI CTS and therefore not incorporated into the PI ITS.
TA	140	Incorporated TSTF-36, Rev. 4.
	141	Not used.
PA	142	NUREG-1431 SR 3.8.2.1 and associated Bases have been revised by making the SR numbers in the Notes and the SR to be consistent with the SRs in ITS 3.8.1. Some of the SRs identified in SR 3.8.2.1 have been deleted or renumbered in LCO 3.8.1. Reference specific justifications for LCO 3.8.1.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	143	NUREG-1431 Bases 3.8.3, Background Section is being revised by deleting the following, "...for a period of 7 days while the DG is supplying maximum post loss of coolant accident load demands....", and "The maximum load demand is calculated using the assumption that a minimum of any two DGs is available." These statements are not accurate relative to the PI CLB. The maximum load demand, associated calculations, and assumptions are discussed in detail in the PI USAR. This change is considered to be administrative in nature providing consistency with the PI USAR.
	144	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	145	NUREG-1431 LCO 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, and associated Bases have been modified by deleting all references to lube oil and starting air. This results in the deletion or revision of various ISTS Conditions, Required Actions, and SRs as noted in the ITS markup document. The requirement to ensure lube oil inventory (ISTS Condition B) is maintained is designed for the DGs which consume lube oil during operation, in order to verify that the lube oil available will support 7 days of operation. The DGs at PI do not consume lube oil during operation in an amount significant to be quantified. Hence, the lube oil inventory is not expected to change during operation. Lube oil level is verified by sight glass indication prior to each normal DG start and when placing a DG in the standby condition in accordance with station procedures. Therefore, adequate means are presently used to verify lube oil inventory and it is justifiable to delete such references from this specification. The requirement to ensure starting air pressure (ISTS Condition E) is within limits and provide explicit action requirements for low starting air pressure, is not an explicit requirement in the CTS or CLB. Along with routine DG standby readiness monitoring, installed DG low air pressure alarms, and the definition of OPERABILITY, adequate assurance of proper maintenance of starting air pressure and necessary actions on low air pressure is provided without explicit TS reference.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	146	NUREG-1431 LCO 3.8.3, Conditions C, D, and portions of F, and associated Bases Sections have been revised or deleted. ISTS Condition C (PI Condition B) requires that with one or more DGs with stored fuel particulates not within limits, restore the fuel to within limits within 7 days. This has been revised to state, "Required DG fuel oil tank with stored fuel oil properties not within limits, restore the fuel oil tank to within limits within 7 days." This Condition is only entered when SR 3.8.3.2 fails to meet its acceptance criterion. PI Diesel Fuel Oil Testing Program specifies the properties and associated limits for new and stored fuel oil. ISTS Condition C only specifies total particulates to be within limits which is not completely accurate nor does it adequately reflect the PI Diesel Fuel Oil Testing Program. The PI Diesel Fuel Oil Testing Program specifies other parameters or properties to be monitored and tracked. Therefore, replacing the terms "total particulates" with "properties" is more accurate and applicable to PI. The Bases have been revised to reflect this change. In addition the Bases have been revised deleting the following statement, "Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend." This statement does not add

Difference Category	Difference Number	Justification for Differences
CL	3.8-146	(continued)

any significance or direction, and is not specifically in compliance with current PI practices.

ISTS Condition D requires that if one or more DGs with new fuel oil properties not within limits, restore the new fuel oil to within limits within 30 days. Oil quality is monitored by the PI Diesel Fuel Oil Testing Program and existing plant procedures which contain oil quality property limits. If fuel oil is not within limits, the Fuel Oil Storage Tank is declared out of service and action initiated to restore it to within limits. In addition, each PI unit is designed with a separate and independent new fuel oil receiving tank. When new fuel oil is received, it is placed in the fuel oil receiving tank where it is tested. Once the test results are received, verifying that the fuel oil properties are within limits, it is then transferred to the fuel oil storage tanks where it can be used for the DGs. PI CTS does not require this specific Required Action again, based on the design of the fuel oil system.

ITS Condition C has been added stating, "Required Action and associated Completion Time of Condition B not met, isolate the associated DG fuel oil tank immediately." If a fuel oil tank is not restored to within limits after 7 days, the tank is isolated and cannot be used for the DGs. If after isolating the tank, there is not sufficient fuel oil available, as required by ITS Condition A, ISTS Condition F (PI Condition D) is

Difference Category	Difference Number 3.8-	Justification for Differences
CL	146	(continued) entered. During isolation of the tank, the fuel oil in the tank may be replaced or brought back within limits. Once the stored fuel oil is restored to within limits, the tank can be unisolated and returned to service. ISTS Condition F has been revised by adding the minimum stored DG fuel oil quantity limits that can support the PI safety analysis. If the stored DG fuel oil falls below the specified quantity, the DG(s) is to be declared inoperable immediately. The addition of this new requirement ensures that PI will have adequate supply of stored DG fuel oil for it to perform its intended safety function for the analyzed amount of time.
CL	147	NUREG-1431 SR 3.8.3.5 and associated Bases Section have been deleted. Water in the fuel oil storage tanks is tested in SR 3.8.3.2. PI has not had a history of water accumulation in the fuel oil tanks and therefore, water does not have to be routinely removed. In addition, any testing requirements for fuel oil are contained in the PI Diesel Fuel Oil Testing Program.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	148	NUREG-1431 Bases 3.8.4, Condition A (PI Condition C) has been revised by deleting the first paragraph. This paragraph is not consistent with or applicable to LCO Condition C.
	149	Not used.
PA	150	NUREG-1431 Bases 3.8.3, Background Section has been revised. The following sentence has been deleted, "All outside tanks. and piping are located underground." This statement is not accurate with the PI design. PI design currently has some tank vent and fill piping above the ground. This design has been approved during the initial licensing of PI. Removal of this information and does not change or alter any technical requirements or the way any equipment is operated.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	151	NUREG-1431 Bases 3.8.3, Background Section has been revised by deleting the following, "Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3)." PI is not committed to the subject Regulatory Guide; therefore, any reference or implication that PI is in compliance with the subject Regulatory Guide is being deleted. In addition, the statement, "... SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level." has been deleted from the ISTS Bases. The PI Diesel Fuel Oil Testing Program, as required by ITS 5.0, provides the requirements for the DG fuel oil testing. These changes are consistent with the CTS, CLB, and the PI Diesel Fuel Oil Testing Program.
CL	152	NUREG-1431 Bases 3.8.3, LCO Section stated, " ... sufficient supply for 7 days of full load operation." This sentence has been changed to state, " ... sufficient supply for one DG on each unit to operate for 14 days (Ref. 1)." Bases 3.8.3, Condition A, is edited to incorporate 14 day and 12 day limits, based on the ISTS provision of 7 day and 6 day limits. This change is consistent with PI design and the Background and SR 3.8.3.1 Section of the ITS.

Difference Category	Difference Number 3.8-	Justification for Differences
	153	Not used.
CL	154	NUREG-1431 Bases 3.8.3, Background Section has been revised providing additional information and clarification about the design and operations of the DG fuel oil receiving tank, the safeguards fuel oil storage tanks, and the PI Diesel Fuel Oil Testing Program. New fuel oil is placed in the DG fuel oil receiving tank where it is tested in accordance with the PI DG Fuel Oil Testing Program. Once the test results have been certified that the fuel oil is within limits, the fuel oil is transferred, using the DG tank pump, to one of the fuel oil storage tanks (4 for each unit). This is additional information and does not change or alter any technical requirements or the way any equipment is operated.
CL	155	NUREG-1431, Rev. 1, Bases 3.8.3 has been revised throughout deleting all diesel fuel oil testing requirements and statements not specifically applying to PI. To make the Bases read correctly, it has been updated throughout reflecting PI CLB and CTS requirements for testing DG fuel oil.
TA	156	Incorporated TSTF-2, Rev. 1.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	157	NUREG-1431 Bases 3.8.2, Required Action A has been revised making it more consistent with PI design and current outage practices. The Required Action currently implies that the required path of offsite power is to be available to all trains. This has been revised to clarify that the required path needs to be available to at least one of the required trains. During the shutdown conditions of LCO 3.8.2, not all (both) trains are required to be OPERABLE.
PA	158	NUREG-1431 Bases 3.8.6, Actions Section has been revised by adding a discussion describing the Note which applies to the Actions of this LCO. The Note adds the flexibility of separate condition entry.
PA	159	NUREG-1431 LCO 3.8.6, Required Action A.3 Completion Time has been revised from 31 days to 72 hours. The 72 hours is reasonable to restore the battery cell voltage due to the time required, especially if this condition is identified during non-normal working hours, thus bringing in appropriate personnel from offsite and performing the restoration. Due to the low probability of an event occurring during this period, the 72 hours needed to restore the cell voltage is acceptable.

Difference Category	Difference Number 3.8-	Justification for Differences
TP	160	Incorporated industry traveler TSTF-360. TSTF-360 has been revised in several places to be consistent with PI CLB and design.
PA	161	NUREG-1431 LCO 3.8.6 Condition B (PI Condition E) was revised by deleting the following, "OR One or more batteries with average electrolyte temperature of the representative cells < 60°F." This Condition is covered under PI ITS Condition D and is therefore not needed to be stated in this Condition.
PA	162	NUREG-1431 Bases 3.8.6, Background Section has been revised by TSTF 360. The Traveler has been revised in order to make it applicable to PI . Specifically, the first paragraph in the second sentence states, "The battery cells are of flooded lead acid construction with a nominal specific gravity of [1.215]." This has been revised to state," ... specific gravity as required by the manufacturer." PI batteries are manufactured by two different companies. Putting the specific gravity values for each unit would not provide any valuable information. The specific gravity values are contained in the appropriate procedures. By noting that the specific gravity values are in accordance with the manufacturers' requirements is adequate and consistent with other Background Section discussions in the ITS.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	163	NUREG-1431 Bases 3.8.1, 3.8.4, 3.8.6, and 3.8.9 Applicable Safety Analyses Section have been revised by deleting, "or all onsite AC power". PI Safety Analysis for this system does not assume a loss of all onsite power. Therefore, this statement is deleted to be consistent with PI Safety Analysis.
	164	Not used.
CL	165	NUREG-1431 LCO 3.8.9, Conditions A, B, and C Completion Times and associated Bases have been revised. The Completion Time limit of "16 hours from the discovery of failure to meet the LCO" was not included in the ITS since PI does not have this requirement in the CTS. The intent of adding this limit to the Completion Time is to prevent a plant from continuously being in the LCO without ever meeting the full LCO requirements. This abuse of the LCO can be adequately addressed in plant procedures. Including this statement in the ITS would only add confusion for the operators.
	166	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	167	NUREG-1431 Bases 3.8.9, has been revised throughout providing additional information or deleting detail in the Bases to make them more applicable to PI design, operations, and testing. For example, in the LCO Section, deleted the paragraph discussing tie breakers between redundant trains since PI design does not include tie breakers between the trains.
PA	168	NUREG-1431 SR 3.8.10.1 has been revised by adding the following, "Verify correct breaker "and switch" alignments ... ." Adding switches makes this a more accurate SR for the PI design. PI has both breakers and switches in the safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems. This change is consistent with PI design and current operating practices.
PA	169	NUREG-1431 Bases 3.8.9, Action C.1, is being revised by deleting the following sentence, "The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3)." This sentence is being deleted because it is not referenced in the subject Regulatory Guide; in addition, reference JFD CL3.8-172 for PI position on Regulatory Guides.

Difference Category	Difference Number 3.8-	Justification for Differences
	170	Not used.
CL	171	NUREG-1431 LCO 3.8.4, Required Actions A.2, B.1, and C.1 Completion Times and their associated Bases have been revised to 8 hours to be consistent with CLB as in the PI CTS. Maintaining CLB was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement.
CL	172	NUREG-1431 Bases 3.8 has been revised deleting references to specific Regulatory Guides, IEEE Standards, and 10 CFR 50 criteria that PI is either not committed or designed to. PI was designed, built, and licensed prior to 10 CFR 50 Appendix A GDC and other noted NRC/industry design criteria. Where specific Industry Standards or Regulatory Guides are referenced, within the ITS, it does not mean PI is committing to them. They are only used as reference to support the ITS Bases or NRC criteria, Frequencies, SRs, etc., that are consistent with PI CLB.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	173	NUREG-1431 Bases 3.8.4, Background Section, has been revised by changing the information about the "spare" battery charger to be applicable to the PI "portable" charger design and usage. At PI, there is a portable battery charger that may be moved into place to be used in either unit. This portable charger has been approved in the PI initial SER.
PA	174	NUREG-1431 LCO 3.8.6 and Applicability statements have been revised. LCO 3.8.6 as currently revised by TSTF 360 states, "Battery parameters for Train A and B batteries shall be within limits." This statement has been further revised by deleting the following statement, " ... for Train A and Train B batteries ... ." Since PI only has Train A and Train B safeguards batteries, the deleted statement does not provide any significance or clarification. In addition, the APPLICABILITY statement currently states, " When associated DC electrical power subsystems are required to be OPERABLE." This has been revised to state, " When the battery is required to be OPERABLE." This revision provides clarification to the current APPLICABILITY statement. These revisions do not change any technical intent of the ISTS; they only provide clarification as applicable to PI.

Difference Category	Difference Number 3.8-	Justification for Differences
TA	175	NUREG-1431 LCO 3.8.5, 3.8.8, and associated Bases have been revised consistent with the guidance of TSTF-204, Rev. 3.
PA	176	NUREG-1431 Bases 3.8.8, LCO Section has been revised by deleting the first sentence. This is consistent with TSTF-204, Rev. 3, which clarifies that safety analyses for Shutdown MODES operation does not consider Operating DBA's. The sentence is not consistent with PI CLB since PI does not currently have Shutdown Technical Specifications.
CL	177	NUREG-1431 Bases 3.8.5 and 3.8.8 have been revised providing additional information and clarification consistent with PI design, terminology, and operating practices, since PI does not currently have Shutdown Technical Specifications. This clarification takes TSTF-204, Rev. 3, into account.
	178	Not used.
	179	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	180	NUREG-1431 Bases 3.8.7, LCO Section has been revised by deleting the following statement, "... output voltage and frequency within tolerances, and ... ." SR 3.8.7.1 does not require either the voltage or the frequency to be verified or be within any tolerances. Therefore, this statement is inconsistent with the requirements of the Specification and deleted.
	181	Not used.
	182	Not used.
CL	183	NUREG-1431 LCO 3.8.7, Required Action A.1 Completion Time and associated Bases have been decreased from 24 hours to 8 hours. The decrease in Completion Time to 8 hours is consistent with the CTS. Maintaining CLB was agreed to be acceptable between the industry and NRC during the onset of the ITS conversion project. This change is consistent with that agreement.
	184	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	185	NUREG-1431 Bases 3.8.7 LCO Section contains an explanation of the Note which allows an instrument bus inverter to be disconnected from its associated DC bus for up to 24 hours while performing an equalizing charge on the battery. The inverters used at PI are not required to be disconnected during equalizing charges. Therefore, this Note has been deleted consistent with ITS.
PA	186	NUREG-1431 Bases 3.8.4 Background Section, has been revised by deleting the following, "The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand." This information is contained in the USAR and does not need to be incorporated into the ITS. In addition, the reference to Train A and Train B was deleted since PI only has two Trains.
	187	Not used.
	188	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
	189	Not used.
PA	190	NUREG-1431 LCO 3.8.10, Required Action A.2.5 states, " Declare associated required residual heat removal subsystem(s) inoperable and not in operation. " This statement has been revised by deleting, " ... and not in operation." If the pump is inoperable, PI considers it to be not in operation. Therefore, this phrase is not required.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	191	NUREG-1431 Bases 3.8.10, Background Section has been revised by adding a discussion about the use of "alternate" power distribution equipment during MODES 5 and 6. During the unit shutdown, PI has available the safeguards AC, DC, and reactor protection instrument AC electrical power distribution systems. These power distribution systems feed various plant equipment required to be OPERABLE during plant shutdown conditions. However, during an outage, maintenance and testing is required to be done on parts of the above distribution systems, therefore making them inoperable. In order to maintain electrical power to required systems, PI also uses the following alternate power distribution equipment; 4kV bus tie, 480kV alternate feeds, Panel 117 (217 for unit 2), 117 to 217 cross tie, and the Service Building DC Safeguards DC cross tie. The alternate power distribution equipment provides a reliable power supply to the various plant systems or equipment required to be OPERABLE during MODES 5 and 6. This change is consistent with CLB and current plant practices.
PA	192	NUREG-1431 Bases 3.8.5, 3.8.8, and 3.8.10, Applicable Safety Analyses have been replaced with the Applicable Safety Analyses discussion from Bases 3.8.2. Bases 3.8.2 provides additional information which is applicable and better explained than in the other Sections.

Difference Category	Difference Number 3.8-	Justification for Differences
	193-196	Not used.
CL	197	NUREG-1431 LCO SR 3.8.1.5 and associated Bases have been deleted. PI day tanks are not designed with any type of drain in the tank that would allow draining any water. PI operating history has shown that the day tanks have not had any water accumulation problems. In addition, neither PI CTS or CLB require checking the day tanks for water; therefore, this SR is being deleted.
	198	Not used.
	199	Not used.
CL	200	NUREG-1431 Bases 3.8 has been revised to reflect current PI design and operating practices. As an example, Bases 3.8.1, Required Action B.2 states, "This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included." PI has two 100% capacity auxiliary feedwater pumps, a motor and a turbine driven. The turbine driven auxiliary feedwater pump is not supported by the DG. Therefore, this statement is not applicable to PI design and is deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	201	NUREG-1431 Bases 3.8.1, Background Section has been revised by adding the statement, "... the Unit 1 DGs meet the intent of Safety Guide 9 and Unit 2 DGs satisfy the intent of Regulatory Guide 1.9, ..." This statement was added to reflect the differences between the two unit DGs. Unit 1 DGs were installed prior to the issuance of Regulatory Guide 1.9. Therefore, Unit 1 DGs rating were consistent with Safety Guide 9. When Unit 2 DGs were installed, Regulatory Guide 1.9 has been issued; however, PI did not adopt this Regulatory Guide in its entirety as discussed in the PI USAR. This change is consistent with the PI CLB.
CL	202	NUREG-1431 Bases 3.8.1 and 3.8.2, LCO Section have been revised by replacing the statement, "This will be accomplished ... ." with "The DG will be ready to load ... following receipt of a start signal." PI design is that each DG is capable of starting, accelerating to the required speed and voltage, and ready to be loaded within 10 seconds. PI DGs are not required to be loaded within 10 seconds. In addition, Bases 3.8.2, LCO statement has been revised by deleting the statement, "This sequence must be accomplished within [10] seconds." As stated above, the PI DGs are required to be ready to load within 10 seconds upon receipt of a start signal. Therefore, the Bases is revised to reflect the PI design and CLB.

Difference Category	Difference Number 3.8-	Justification for Differences
	203	Not used.
PA	204	NUREG-1431 Bases 3.8.1, LCO Section has been revised by deleting the subject paragraphs. The subject paragraphs discuss various information about the AC sources in a train and the AC offsite sources being independent and separated to the extent practical. PI USAR provides a detailed discussion about the design of the AC trains and offsite sources; therefore, this redundant information is not needed in the TS and is being deleted. Also, Bases SR 3.8.1.1 has been revised by editing the sentence discussing preferred power source. PI design does not identify a preferred power source. As above, the USAR provides a detailed discussion about the offsite sources.
PA	205	NUREG-1431 Bases 3.8.1, 3.8.4, 3.8.7, and 3.8.9 Applicability Sections have been revised by deleting the following, " ...or abnormal transients;" PI considers an abnormal transient as an AOO. Therefore, the specific reference to an abnormal transient is being deleted.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	206	NUREG-1431 Bases 3.8.1, Condition C is for two paths inoperable. Required Action C.1 states to declare required feature(s) inoperable when its redundant required feature(s) is inoperable with a Completion Time of 12 hours. The ISTS states that the justification for the 12 hours is Regulatory Guide 1.93. PI CTS already has a Completion Time of 12 hours. Therefore, any references in the ISTS to the Completion Time being shorter or reduced is deleted.
CL	207	NUREG-1431 Bases 3.8.1, Required Action C.1 and C.2 have been revised by deleting the subject discussions since they are referring to Regulatory Guide 1.93. Since PI is not committed to Regulatory Guide 1.93, the subject discussions are not applicable to PI.
PA	208	NUREG-1431 Bases 3.8 has been revised deleting redundant information that also appears in the USAR.
	209	Not used.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	210	NUREG-1431 Bases 3.8.2, LCO statement has been revised deleting the following, "It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains." PI design does not provide a cross tie between the trains. The design, as described in the USAR, provides for each offsite source being capable of supplying both trains, but this not termed a cross tie.
PA	211	NUREG-1431 LCO 3.8.2 and associated Bases has been revised by adding a Note allowing the LCO not being applicable for a period of 8 hours during the performance of SR 3.8.1.10. Without the Note, the LCO requires that one DG capable of supplying one train of the onsite 4 kV safeguards distribution system required by LCO 3.8.10 be OPERABLE. SR 3.8.2.1 requires the SRs of Specification 3.8.1 be performed at their specified Frequencies for those AC sources that are required to be OPERABLE to support those systems operating during plant shutdown. One of these SRs requires DG testing. At PI, when a DG is being tested, and thus operating, it is considered to be inoperable since during this testing some controls must be placed in manual. SR 3.8.1.10 in particular results in considering both DGs inoperable during test performance. The 8 hour period is reasonable to allow performance of the required SR.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	212	NUREG-1431 Bases 3.8.5 Background Section has been revised by adding the following information, "In addition to the safeguards AC, the service building battery and or charger may be used as alternate sources during plant shutdown. These alternate sources may be considered to be a required power source available to provide reliable power to various plant systems and equipment that are required to be OPERABLE to support shutdown conditions." This additional information is provided to more accurately discuss the PI design and use of the power sources.
PA	213	NUREG-1431 LCO 3.8.9, Required Actions A, B, and C have been revised by adding the following, "Declare associated required supported feature(s) inoperable, Immediately." This Action needed to be added to provide guidance for when a portion of safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems is inoperable. This condition is not covered in the ISTS. This change is consistent with the practices at PI.

Difference Category	Difference Number 3.8-	Justification for Differences
CL	214	NUREG-1431 LCO 3.8.9 Required Action E has been revised by adding the following, "Two or more Reactor Protection Instrument AC Panels inoperable, Enter LCO 3.0.3, Immediately." This Required Action has been added to provide specific Actions when two or more Reactor Instrument AC panels are inoperable, since the instrument AC panels are distinct from "Two trains...". The ISTS does not currently specify this condition.
CL	215	NUREG-1431 SR 3.8.5.1 and associated Bases has been revised to be consistent with the associated SRs in ITS 3.8.4.
CL	216	NUREG-1431 Bases 3.8.7, Background Section has been revised by deleting the following sentence, "Specific details on inverters and their operating characteristics are found in the USAR." This statement is being deleted because the PI USAR does not contain this detailed information.

Difference Category	Difference Number 3.8-	Justification for Differences
PA	217	NUREG-1431 Bases 3.8.7 and 3.8.8, Applicable Safety Analyses Section has been revised by changing the last sentence to be consistent with the rest of the ISTS.

PACKAGE 3.8

ELECTRICAL POWER SYSTEMS

PART G

NO SIGNIFICANT HAZARDS DETERMINATION (NSHD)

and

ENVIRONMENTAL ASSESSMENT

for

CHANGES TO PRAIRIE ISLAND CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Part G

PACKAGE 3.8

ELECTRICAL POWER SYSTEMS

NO SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10CFR Part 50, Section 50.91 using the standards provided in Section 50.92.

For ease of review, the changes are evaluated in groupings according to the type of change involved. A single generic evaluation may suffice for some of the changes while others may require specific evaluation in which case the appropriate reference change numbers are provided.

A - Administrative (GENERIC NSHD)

(A3.8-01, A3.8-10, A3.8-13, A3.8-15, A3.8-17, A3.8-19, A3.8-20, A3.8-22, A3.8-25, A3.8-30, A3.8-35, A3.8-38, A3.8-39, A3.8-40, A3.8-51, A3.8-53, A3.8-54, and A3.8-56)

Most administrative changes have not been marked-up in the CTS, and may not be specifically referenced to a discussion of change (DOC). This NSHD may be referenced in a discussion of change by the suffix "A" if the change is not obviously an administrative change and requires an explanation.

These proposed changes are editorial in nature. They involve reformatting, renaming, renumbering, or rewording of existing TS to provide consistency with NUREG-1431 or conformance with the Writer's Guide, change of current plant terminology to conform to NUREG-1431 or change of NUREG-1431 terminology to conform to CTS. Some administrative changes involve relocation of requirements within the TS without affecting their technical content. Clarifications within the NEW PI ITS which do not impose new requirements on plant operation are also considered administrative.

A - Administrative (continued)

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes involve reformatting, renumbering and rewording of the existing Technical Specifications, along with other Technical Specification changes discussed above. The reformatting, renumbering, rewording, and other changes involved no technical changes to the existing Technical Specifications, and were done in order to be consistent with the NUREG-1431, Rev. 1. During the development of NUREG-1431, certain wording preferences or English language conventions were adopted. The proposed changes are administrative in nature, and do not impact initiators of analyzed events. The analyzed events are initiated by the failure of plant structures, systems or components (SSCs). Because these changes do not impact the condition or performance of these SSCs, the probability of the occurrence of an analyzed event is not significantly increased.

   The consequences of the analyzed events are the result of the plant being operated within assumed parameters at the onset of an event, and the successful functioning of at least one train or division of the credited equipment to mitigate the event. Because of the administrative nature of these changes, there is no impact on the capability of the credited equipment to perform, nor is there an influence on the likelihood that credited equipment will fail to perform.

   The operation of the plant within the assumed parameters is governed by the technical aspects of the Technical Specifications, and these administrative changes do not impact these technical requirements. As such, there is no significant increase in the consequences associated with analyzed events.

   Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

A - Administrative (continued)

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. No setpoints for parameters which initiate protective or mitigative action are being changed. As a result, no new failure modes are being introduced. There are no changes in the procedures or methods governing normal plant operation, nor are the procedures utilized to respond to plant transients altered as a result of the administrative changes. The changes do not impose any new or different requirements or eliminate any assumptions made in the safety analysis, nor do they impact the licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through the design of the plant SSCs, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed changes are administrative in nature and do not involve any technical changes. As such, they do not impact any safety analysis assumptions and no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

M - More restrictive (GENERIC NSHD)
(M3.8-04, M3.8-06, M3.8-14, M3.8-18, M3.8-21, M3.8-27, M3.8-31, M3.8-41, M3.8-42, M3.8-47, M3.8-48, M3.8-49, M3.8-50, M3.8-52 and M3.8-55)

This proposed TS revision involves modifying the CTS to impose more stringent requirements upon plant operations to achieve consistency with the guidance of NUREG-1431, correct discrepancies or remove ambiguities from the specifications. These more restrictive TSs have been evaluated against the plant design, safety analyses, and other TS requirements to ensure the plant will continue to operate safely with these more stringent specifications.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes provide more stringent requirements for operation of the plant. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event.

   These more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed changes do not involve any physical alteration of the plant, that is, no new or different type of equipment will be installed, nor do they change the methods governing normal plant operation.

M - More restrictive (continued)

These more stringent requirements do impose different operating restrictions. However, these restrictions are consistent with the assumptions made in the plant safety analyses and licensing bases. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The imposition of more stringent requirements on plant operation either has no impact on the plant margin of safety nor increases the margin of safety. Each change in this category is by definition providing additional restrictions to enhance plant safety by:

a) increasing the analytical or safety limit;
b) increasing the scope of the specifications to include additional plant equipment;
c) adding requirements to current specifications;
d) increasing the applicability of the specification;
e) providing additional actions;
f) decreasing restoration times;
g) imposing new surveillances; or
h) decreasing surveillance intervals.

These changes maintain requirements within the plant safety analyses and licensing bases. Therefore, these changes do not involve a significant reduction in a margin of safety.

R - Relocation (GENERIC NSHD)
(None in this package)

This LAR proposes to relocate requirements contained in the CTS out of the TS into licensee controlled documents. These requirements are relocated because they 1) do not meet the NRC ITS selection criteria defined in 10 CFR 50.36(c)(2)(ii); or 2) are mandated by current NRC regulations and are therefore unnecessary in the TS.

In the NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (dated 7/16/93), the NRC stated:

> . . . since 1969, there has been a trend towards including in Technical Specifications not only those requirements derived from the analyses and evaluations included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power reactors. . . This has contributed to the volume of TS and to the several-fold increase, since 1969, in the number of license amendment applications to effect changes to the TS. It has diverted both staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety.

Thus, relocation of unnecessary requirements from the CTS should result in an overall improvement in plant safety through more focused attention to the requirements that are most important to plant safety.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   These proposed changes relocate requirements for SSCs or variables which do not meet the criteria for inclusion in the improved TS or duplicate regulatory requirements. The affected SSCs or variables are assumed not to be initiators of analyzed events and are assumed not to mitigate accident or transient events.

   These relocated requirements will continue to be maintained pursuant to 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Controls section of the PI ITS.

R - Relocation (continued)

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes do not impose any different requirements and adequate control of existing requirements will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   These proposed changes will not reduce the margin of safety because they do not impact any safety analysis assumptions. In addition, the relocated requirements for the affected SSCs or variables are the same as the CTS. Since future changes to these requirements will be evaluated per the requirements of 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Control section of the ITS, proper controls are in place to maintain the plant margin of safety. Therefore, these changes do not involve a significant reduction in the margin of safety.

LR - Less restrictive, Relocated details (GENERIC NSHD)
(LR3.8-02, LR3.8-34, LR3.8-37, LR3.8-43, LR3.8-44, and LR3.8-45)

Some information in the PI CTS that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been relocated to the proposed Bases, USAR or other licensee controlled documents. The relocation of this descriptive information is acceptable because these documents will be controlled under the controls of 10CFR50.59. Therefore, the descriptive information that has been relocated will continue to be maintained in an appropriately controlled manner.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes relocate detailed, descriptive requirements from the CTS to the PI ITS Bases, USAR or other licensee controlled documents. These documents containing the relocated requirements will be maintained under the provisions of 10CFR50.59, or a program or procedure based on 10CFR50.59 evaluation of changes. Since these documents, to which the CTS requirements have been relocated, are evaluated under 10CFR50.59 or its guidance, no increase in the probability or consequences of an accident previously evaluate will be allowed without prior NRC review and approval. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not necessitate physical alteration of the plant, that is, no new or different type of equipment will be installed, or change parameters governing normal plant operation. The proposed changes will not impose any different requirements and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

LR - Less restrictive, Relocated details (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be relocated from the CTS to the Bases, USAR or other licensee controlled documents are the same as the existing CTS. Since future changes to these requirements will be evaluated under 10CFR50.59 or its guidance, no reduction in a margin of safety will be allowed without prior NRC approval. Therefore, these changes do not involve a significant reduction in a margin of safety.

L - Less restrictive, Specific

Each CTS change which is designated as Less (L prefix) restrictive on plant operations is provided with a specific NSHD.

Specific NSHD for Change L3.8-07

The ITS adds a new requirement in LCO 3.8.1, Condition B.2 which declares required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable. This Condition is not in the CTS; however, its addition does provide some relaxation and therefore is considered to be a Less Restrictive change.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The DGs are not considered as initiators of any analyzed event; therefore, this change has no impact on the probability of an event previously analyzed. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. This change does not affect the performance of any credited equipment. The minimum required equipment to respond to analyzed events is still assured by requiring the equipment supported by the inoperable DG be declared inoperable if its redundant counterpart is inoperable. This provides for actions appropriate to the actual inoperabilities. As a result, no analyses assumptions are violated. The extension in time provides additional time to either restore the inoperable DG or the inoperable feature, while still assuring no loss of safety function exists. This time extension is considered reasonable to complete repairs prior to requiring a forced shutdown of the unit. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

Specific NSHD for Change L3.8-07 (continued)

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes any parameters governing normal plant operation. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed nor is any change being made to the procedures relied upon to respond to an off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis nor licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change, which allows transferring actions to affected equipment (by declaring that affected equipment inoperable), and additional time to restore the inoperabilities while still assuring no loss of safety function, will not result in any detrimental impact of any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.8-09

The following CTS Actions and SRs contain various testing or verification requirements that are not included in the ITS:

CTS 3.7.B.2 requires that with a path inoperable, the associated DG(s) are required to either be running or demonstrated to be OPERABLE. The ITS does not specifically require the DG to be running or starting the DG(s) to demonstrate OPERABILITY.

CTS 3.7.B.3 requires that with one path and one DG inoperable, demonstrate the other DG to be OPERABLE and verify the other path is OPERABLE. The ITS does not specifically require demonstration for the other DG OPERABILITY nor verification of the other path's OPERABILITY.

CTS 3.7.B.4 requires with two paths inoperable, both DGs are either running or are demonstrated OPERABLE. The ITS does not specifically require the DGs to be running nor demonstrating them to be OPERABLE.

CTS 3.7.B.6 requires that if one or more 4 kV safeguards AC electrical power subsystems are inoperable, verify that the 4 kV safeguards bus and its associated 480 V safeguards bus are OPERABLE, and the DGs and safeguards equipment associated with the redundant train are OPERABLE. The ITS provides additional acceptable options for this condition, none of which require the subject CTS verifications.

CTS 3.7.B.7 requires that if one battery charger is inoperable, then the DG and safeguards equipment associated with its counterpart are OPERABLE. The ITS only requires that the associated battery is OPERABLE and does not specifically require that the DG nor the safeguards equipment associated with the other battery be verified to be OPERABLE.

CTS 3.7.B.8 requires that if one DC safeguards electrical power subsystem is inoperable, the other battery and both battery chargers must be OPERABLE. The ITS only requires that the DC safeguards subsystem be restored to OPERABLE and does not specifically require both chargers and the other battery to be OPERABLE.

Specific NSHD for Change L3.8-09 (continued)

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. These changes do not affect the performance of any credited equipment. Even if an inoperable plant SSC condition exists, the minimum required equipment needed to respond to the analyzed events is still assured since other redundant or comparable equipment, if supported by the OPERABLE path(s), battery(ies), battery charger(s), or DG(s) will be able to perform their intended safety function. The redundant or comparable SSCs are part of the "safety in depth" as well as the design of the plant. Eliminating some of the CTS requirements for verifying that specific systems are OPERABLE is not needed since SSCs are considered to be OPERABLE without further verification or testing if ; 1) has passed its last SR, 2) the SRs are current for that SSC, and 3) the SSC is not inoperable for other reasons. This is the basic philosophy of the ITS. In addition, the ITS provides adequate Required Actions for inoperable SSCs, thus eliminating some of the additional testing or verifications required by the CTS. This provides for actions appropriate to the actual inoperabilities. As a result, no analyses assumptions are violated. The extension in time provides additional time to restore either the inoperable DG or the inoperable feature, while still assuring no loss of safety functions exists, and is considered a reasonable time to complete repairs prior to requiring a forced shutdown of the unit. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed. No change is being made to the procedures relied upon to respond to any off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made

Specific NSHD for Change L3.8-09 (continued)

in the safety analysis nor licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident previously evaluated. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available that actuates upon demand for the purpose of mitigating an analyzed event. These changes do not affect the performance of any credited equipment. Even if an inoperable plant SSC condition exists, as stated above, the minimum required equipment to respond to analyzed events is still assured since other redundant or comparable equipment, required by the TSs, supported by the OPERABLE path(s), battery(ies), battery charger(s), or DG(s) will be able to perform their intended safety function. This redundancy or comparable SSCs are part of the "safety in depth" as well as the design of the plant. Eliminating some of the CTS requirements for verifying that specific systems are OPERABLE is not needed since SSCs are considered to be OPERABLE without further verification or testing if: 1) it has passed its last SR; 2) the SRs are current for that SSC; and 3) the SSC is not inoperable for other reasons. This is the basic philosophy of the ITS. In addition, the ITS provides adequate Required Actions for inoperable SSCs, thus eliminating some of the additional testing or verifications required by the CTS. These proposed changes maintain appropriate Required Actions ensuring safe plant operation while in the degraded condition, eliminates excessive testing, and operator actions. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.8-11

CTS requires that two offsite paths from the grid to the 4 kV safeguards distribution system may be inoperable for 12 hours provided the DGs are OPERABLE. The ITS relaxes the Completion Time from 12 hours to 24 hours. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change extends the Completion Time from 12 hours (CTS) to 24 hours when two offsite paths are inoperable. This Completion Time extension is consistent with Regulatory Guide 1.93 which allows operation to continue in this Condition. While in this Condition, the plant maintains onsite AC sources which will provide a level of safety to mitigate any analyzed event, safely shutdown the plant, and maintain it in a safe shutdown condition.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. This change does not affect the performance of any credited equipment. The minimum required equipment to respond to analyzed events is still maintained while in the associated Condition. This provides for actions appropriate to the actual inoperabilities. As a result, no analyses assumptions are violated. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, nor is any installed equipment being operated in a new or different manner. There is no change being made to the parameters in which the plant is operated. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands

Specific NSHD for Change L3.8-11 (continued)

on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed, and no change is being made to the procedures relied upon to respond to an off normal event. As such, no new failure modes are being introduced. The change does not alter any assumptions made in the safety analysis nor licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The 24 hours is acceptable since with both offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. A simultaneous loss of offsite AC source, a LOCA, and a worst case single failure were postulated as a part of a design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to restore one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria. Relaxing the Completion Time to 24 hours does not significantly impact these factors. There are no design changes or equipment performance parameter changes associated with this change. No setpoints are affected, and no change is being proposed in the plant operating limits as a result of this change. Thus, relaxing the Completion Time does not involve a significant reduction in the plant margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.8-12

CTS 3.7.A.5 requires 51,000 gallons of diesel fuel for Unit 1. Based on actual consumption rate analysis, the CTS number should be 42,000 gallons for Unit 1 DGs. In addition, the CTS does not provide any specific Completion Time in the event the total fuel oil quantity falls below the quantities in the CTS. In this condition, both DGs would be considered inoperable. With both DGs inoperable, there is a 2 hour restoration time or the plant will enter LCO 3.0.3. The ITS allows 48 hours to restore the fuel oil quantity to within the quantity limits. This is considered to be a less restrictive change since the ITS allows 48 hours to replenish the fuel oil to within limits prior to declaring the DGs inoperable. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The DGs are not considered as initiators of any analyzed event; therefore, this change has no impact on the probability of an event previously analyzed. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

   This change reduces the CTS requirement for total Unit 1 diesel fuel oil storage from 51,000 to 42,000 gallons of diesel fuel oil. The new value of 42,000 gallons is based on a calculation using actual Unit 1 DG consumption rates when loaded in accordance with the USAR. The 42,000 gallons meets the PI design basis of maintaining an adequate fuel oil supply to operate one Unit 1 DG for 14 days. In addition, this change extends the time limit to replenish the DG fuel oil to within limits prior to declaring the DGs inoperable. The 48 hours allows sufficient time for replacement volume and performing the analyses required prior to the addition of fuel oil to the tank(s). The 48 hours is also sufficient time to replenish the fuel oil volume prior to declaring the DGs inoperable. The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The successful functioning of the DG is not impacted because the required volume of fuel oil to ensure the plant can mitigate any analyzed event is not jeopardized and would be available in the event of an accident while waiting (48 hours) for the arrival of replenishment fuel oil. Operating experience has demonstrated that the 48 hours is sufficient time to receive the replenishment fuel oil on site. Based on this evaluation, there is not a significant increase in the consequences of a previously analyzed event.

Specific NSHD for Change L3.8-12 (continued)

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced nor is any installed equipment being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed, and no change is being made to the procedures relied upon to respond to an off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   This reduces the amount of stored diesel fuel oil based on actual Unit 1 DG consumption rates while increasing the Completion Time to replenish the DG fuel oil volume prior to declaring the DGs inoperable. However, this increased Completion Time is consistent with Regulatory Guide 1.93. The 48 hour extension is sufficient for PI to receive and test the required fuel oil. This 48 hour extension is acceptable based on the remaining capacity is adequate to supply the DGs with enough fuel oil to perform their intended safety function to mitigate those events analyzed in the USAR. There is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within its prescribed limits as identified and analyzed in the USAR. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.8-16

The CTS only allows one 4 kV safeguards bus and one DC safeguards bus to be inoperable for 8 hours. The proposed change would allow one or more electrical power distribution systems to be inoperable for 8 hours.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change will allow more than one AC, DC, or Reactor Protection Instrument AC power distribution subsystem to be inoperable provided a loss of function does not occur. While a loss of offsite power is an analyzed event, the loss of subsystems (where loss of function does not occur) is not an assumed initiator of any previously analyzed accident. Therefore, the change will have no impact on the probability of an accident previously analyzed.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. This change does not affect the performance of any credited equipment. As a result, no analyses assumptions are violated. The change allows an increased time when more than one power distribution subsystem is allowed to be inoperable; however, the proposed Action E ensures that no loss of function will be allowed to exist without an immediate shutdown being initiated. This action ensures that the minimum number of safety systems will remain available to respond to an event. Therefore, based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced nor is any installed equipment being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter

Specific NSHD for Change L3.8-16 (continued)

the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed, and no change is being made to the procedures relied upon to respond to an off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change continues to provide assurance that the appropriate electrical power is available from the electrical power distribution subsystems. Furthermore, the change recognizes that there is a potential for decreased safety if the unit operators attention is diverted from the evaluation and action necessary to restore power to the affected power distribution subsystem. As a result, any reduction in the margin of safety will be insignificant and offset by the benefit gained in reducing unnecessary plant shutdown transients when equivalent compensatory measures exist to ensure no loss of safety function exists. There is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.8-28

The ITS adds two new Notes to ITS SR 3.8.1.2 which allows credit to be taken for the performance of another SR that is performed on a 184 day Frequency. In addition, the Notes allow the performance of a modified DG start test which involves idling and gradual acceleration to synchronous speed. The CTS does not allow for a modified DG start test; therefore requiring the DG to be restarted to perform the other DG test. This allowance is not in the CTS; however, its addition does provide some relaxation and is therefore considered to be a Less Restrictive change.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The DGs are not considered as initiators of any analyzed event; therefore, this change has no impact on the probability of an event previously analyzed. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. Taking credit for a modified DG start test is not allowed in the CTS. Not providing this allowance would require the DG to be restarted in order to perform other DG testing. Allowing credit to be taken for the performance of a modified DG start test would reduce the additional DG startups. In addition, the modified DG test would still prove DG OPERABILITY and ensure that the DG would be able to perform its intended function when required. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which

Specific NSHD for Change L3.8-28 (continued)

ensure the plant remains within analyzed limits are being proposed, and no change is being made to the procedures relied upon to respond to any off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis nor licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident previously evaluated. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change, which allows credit to be taken for a modified DG start test, does not reflect any inability of the DG system to fulfill its function. There is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.8-29

The ITS adds a new Note to ITS SR 3.8.1.3 which states momentary transients may occur for various reasons during loading, unloading, and steady state operation of the DG. These transients are quickly restored to within limits and do not reflect an inability of the DG system to fulfill its intended function. This Condition is not in the CTS; however, its addition does provide some relaxation and therefore is considered to be a Less Restrictive change.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The DGs are not considered as initiators of any analyzed event; therefore, this change has no impact on the probability of an event previously analyzed. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. Momentary transients may occur for various reasons during loading, unloading, and steady state operation of the DG. However, these transients are quickly restored to within limits and do not reflect an inability of the DG system to fulfill its intended function. Therefore, the DG will be able to perform its assumed consequence mitigation role. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed and no change is being made to the procedures relied upon to respond to any off normal event. As such, no new failure modes are being introduced. The change does not alter

Specific NSHD for Change L3.8-29 (continued)

assumptions made in the safety analysis nor licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident previously evaluated. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change, which allows momentary transients for various reasons during loading, unloading, and steady state operation of the DG, does not reflect any inability of the DG system to fulfill its function. There is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.8-36

CTS 4.6.A.3.b and e require that every 18 months, each DG simulate a loss of offsite power in conjunction with a safety injection signal (SI). The ITS provides a relaxation to this requirement in that either a simulated or actual loss of offsite power in conjunction with a SI signal is acceptable in satisfying the SR requirements and frequency. The acceptance of an actual loss of offsite power in conjunction with a SI is not in the CTS; however, its addition does provide some relaxation and therefore is considered to be a Less Restrictive change.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change adds the phrase "an actual or simulated" in reference to the test signals used to actuate the DGs. The CTS wording requires "simulated signal" be used, which does not allow for an actual signal to be applied in meeting the surveillance. The revised wording will allow the plant to take credit for an actual signal to initiate the protective function being surveilled, as well as a simulated signal. These details of testing (i.e., the method of introducing input to the logic) are not considered as initiators of any analyzed event. Therefore, this change has no impact on the probability of an event previously analyzed. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. This change does not affect the performance of any credited equipment. These details of testing (i.e., the method of introducing input to the logic) are not analysis assumptions. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

Specific NSHD for Change L3.8-36 (continued)

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment is operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed, and no change is being made to the procedures relied upon to respond to an off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis nor licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident previously evaluated. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change, which allows changes to the details of testing (i.e., the method of introducing input to the logic) is acceptable because the tests continue to require appropriate confirmation of the assumed function of the system (and thereby assure continued operability), and more accurately presents acceptable testing conditions. There is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.8-46

CTS 4.6.B.5 requires that the integrity of the Station Battery fuses shall be checked daily. This CTS SR is being deleted based on PI design and taking credit for PI ITS SR 3.8.4.1 on a 7 day Frequency. This change provides some relaxation and therefore is considered to be a Less Restrictive change.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. Both the CTS and ITS require battery OPERABILITY. In order for the battery to be OPERABLE, the fuses in the circuitry must be intact and not blown. CTS requires that the fuses be checked daily when the battery charger is running. The ITS does not require this specific testing; however, it does require that the battery voltage be verified to be within limits every 7 days. If the subject fuses are blown, based on PI design, an alarm in the control room would alert the operators that the fuse(s) were blown. In addition, if the fuses were blown, the battery would not recharge and therefore not pass the 7 day SR as required by the ISTS. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. There are no setpoints, at which protective or mitigative actions are initiated, that are affected by this change. This change will not alter the manner in which equipment is operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures which ensure the plant remains within analyzed limits are being proposed, and no change is being made to the procedures relied upon to respond to an off normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis nor licensing basis. Therefore, the

Specific NSHD for Change L3.8-46 (continued)

change does not create the possibility of a new or different kind of accident previously evaluated. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change only eliminates the specific requirement of performing a daily check of the Station Battery integrity. Based on PI design and incorporated ISTS SRs, battery integrity is maintained and would not be degraded in any way by the deletion of this SR. This change will not result in any detrimental impact or any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

The Nuclear Management Company has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.

PACKAGE 3.8

ELECTRICAL POWER SYSTEMS

CROSS - REFERENCE

CURRENT TECHNICAL SPECIFICATIONS

TO

IMPROVED TECHNICAL SPECIFICATIONS

List of Section Cross - References

3.7
4.6

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.7				
3.7.A		LCO	3.8.1	
3.7.A		LCO	3.8.4	
3.7.A		LCO	3.8.7	
3.7.A		LCO	3.8.9	
3.7.A.1		LCO	3.8.1	
3.7.A.2		LCO	3.8.9	
3.7.A.2		(Partial)	Relocated - Bases Table	
New		LCO	3.3.4.c	
3.7.A.3		LCO	3.8.9	
3.7.A.3		(Partial)	Relocated - Bases Table	
3.7.A.4		LCO	3.8.9	
3.7.A.4		(Partial)	Relocated - Bases Table	
3.7.A.5.a		LCO	3.7.8	
3.7.A.5.a		LCO	3.8.1	
3.7.A.5.a		LCO	3.8.3	
3.7.A.5.a		(Partial)	Relocated - Bases	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.7.A.5.b		LCO	3.8.1	
3.7.A.5.b		LCO	3.8.3	
3.7.A.5.b		(Partial)	Relocated - Bases	
3.7.A.6		LCO	3.8.4	
3.7.A.7		LCO	3.8.7	
3.7.A.7		(Partial)	Relocated - TRM	
New		LCO	3.8.2	
New		LCO	3.8.5	
New		LCO	3.8.6	
New		LCO	3.8.8	
New		LCO	3.8.10	
3.7.B		LCO	3.8.1	
3.7.B		LCO	3.8.4	
3.7.B		LCO	3.8.7	
3.7.B		LCO	3.8.9	
3.7.B.1		LCO	3.8.1	
3.7.B.1		SR	3.8.1.2	
3.7.B.2		LCO	3.8.1	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.7.B.3		LCO	3.8.1	
3.7.B.4		LCO	3.8.1	
New		LCO	3.8.1	
3.7.B.5		LCO	3.8.1	
New		LCO	3.8.3	
3.7.B.6		LCO	3.8.9	
3.7.B.6		(Partial)	Relocated - Bases	
3.7.B.7		LCO	3.8.4	
3.7.B.8		LCO	3.8.4	
3.7.B.9			Relocated - TRM	
New		LCO	3.8.3	
New		LCO	3.8.7	
New		LCO	3.8.9	
3.7.B Note*		LCO	3.8.1	
3.7.B Note**			Deleted	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 4.6				
4.6.A.1.a		SR	3.8.1.4	
4.6.A.1.b		SR	3.8.3.1	
4.6.A.1.c		SR	3.8.3.2	
4.6.A.1.d		SR	3.8.1.5	
4.6.A.1.e		SR	3.8.1.2	
4.6.A.1.e		SR	3.8.1.3	
New		SR	3.8.1.1	
New Notes		SR	3.8.1.2 Notes	
New Notes		SR	3.8.1.3 Notes	
4.6.A.2.a		SR	3.8.1.6	
4.6.A.2.b			Relocated - TRM	
4.6.A.2.c		SR	3.8.1.6	
4.6.A.3.a			Relocated - TRM	
4.6.A.3.b.1		SR	3.8.1.10	
4.6.A.3.b.2		SR	3.8.1.10	
4.6.A.3.b.3			Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.6.A.3.c		SR	3.8.1.9	
4.6.A.3.c		(Partial)	Relocated - Bases	
4.6.A.3.d		SR	3.8.1.7	
4.6.A.3.e		SR	3.8.1.8	
New		SR	3.8.2.1	
New		SR	3.8.5.1	
New		SR	3.8.6.1	
New		SR	3.8.6.5	
New		SR	3.8.6.6	
New		SR	3.8.8.1	
New		SR	3.8.9.1	
New		SR	3.8.10.1	
4.6.B.1		SR	3.8.6.2	
4.6.B.1		SR	3.8.6.4	
4.6.B.1		(Partial)	Relocated - TRM	
New		SR	3.8.4.1	
4.6.B.2		SR	3.8.6.3	
4.6.B.2		(Partial)	Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
4.6.B.3			Relocated - TRM	
4.6.B.4		SR (Notes)	3.8.4.3	
4.6.B.4		(Partial)	Relocated - TRM	
4.6.B.5			Relocated - TRM	
New		SR	3.8.4.2	
New		SR	3.8.4.3	
New		SR	3.8.7.1	
4.6.C		SR	3.4.9.2	
4.6.C		(Partial)	Relocated - Bases	

PACKAGE 3.8

ELECTRICAL POWER SYSTEMS

CROSS - REFERENCE

IMPROVED TECHNICAL SPECIFICATIONS

TO

CURRENT TECHNICAL SPECIFICATIONS

Section Cross - Reference

Section 3.8

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
ITS Section 3.8				
3.8.1		LCO	3.7.A	
3.8.1		LCO	3.7.A.1	
3.8.1		LCO	3.7.A.5.a	
3.8.1		LCO	3.7.A.5.b	
3.8.1		LCO	3.7.B	
3.8.1		LCO	3.7.B.1	
3.8.1		LCO	3.7.B.2	
3.8.1		LCO	3.7.B.3	
3.8.1		LCO	3.7.B.4	
3.8.1		LCO	3.7.B.5	
3.8.1		LCO	New	
3.8.1.1		SR	New	
3.8.1.2		SR	3.7.B.1	
3.8.1.2		SR	4.6.A.1.e	
3.8.1.2 Notes		SR	New Notes	
3.8.1.3		SR	4.6.A.1.e	
3.8.1.3 Notes		SR	New Notes	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.8.1.4		SR	4.6.A.1.a	
3.8.1.5		SR	4.6.A.1.d	
3.8.1.6		SR	4.6.A.2.a	
3.8.1.6		SR	4.6.A.2.c	
3.8.1.7		SR	4.6.A.3.d	
3.8.1.8		SR	4.6.A.3.e	
3.8.1.9		SR	4.6.A.3.c	
3.8.1.10		SR	4.6.A.3.b.1	
3.8.1.10		SR	4.6.A.3.b.2	
3.8.2		LCO	New	
3.8.2.1		SR	New	
3.8.3		LCO	3.7.A.5.a	
3.8.3		LCO	3.7.A.5.b	
3.8.3		LCO	New	
3.8.3.1		SR	4.6.A.1.b	
3.8.3.2		SR	New	
3.8.4		LCO	3.7.A	
3.8.4		LCO	3.7.A.6	
3.8.4		LCO	3.7.B	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.8.4		LCO	3.7.B.7	
3.8.4		LCO	3.7.B.8	
3.8.4.1		SR	New	
3.8.4.2		SR	New	
3.8.4.3		SR	New	
3.8.4.3		SR (Notes)	4.6.B.4	
3.8.5		LCO	New	
3.8.5.1		SR	New	
3.8.6		LCO	New	
3.8.6.1		SR	New	
3.8.6.2		SR	4.6.B.1	
3.8.6.3		SR	4.6.B.2	
3.8.6.4		SR	4.6.B.1	
3.8.6.5		SR	New	
3.8.6.6		SR	New	
3.8.7		LCO	3.7.A	
3.8.7		LCO	3.7.A.7	
3.8.7		LCO	3.7.B	
3.8.7		LCO	New	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.8.7.1		SR	New	
3.8.8		LCO	New	
3.8.8.1		SR	New	
3.8.9		LCO	3.7.A	
3.8.9		LCO	3.7.A.2	
3.8.9		LCO	3.7.A.3	
3.8.9		LCO	3.7.A.4	
3.8.9		LCO	3.7.B	
3.8.9		LCO	3.7.B.6	
3.8.9		LCO	New	
3.8.9.1		SR	New	
3.8.10		LCO	New	
3.8.10.1		SR	New	

ITS PACKAGE CONTENTS

Package:

3.9

1. Part A Introduction
2. Part B Proposed PI ITS and Bases
3. Part C Markup of PI CTS
4. Part D DOC to PI CTS
5. Part E Markup of ISTS and Bases
6. Part F JD from ISTS
7. Part G NSHD for changes to PI CTS
8. Cross-Reference CTS to ITS
9. Cross-Reference ITS to CTS

PACKAGE 3.9

REFUELING OPERATIONS

PART A

INTRODUCTION

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

LICENSE AMENDMENT REQUEST DATED December 11, 2000
Conversion to Improved Standard Technical Specifications

3.9
PART A

Introduction to the Discussion of the proposed Changes to the Current Technical Specifications, Justification of Differences from the Improved Standard Technical Specifications, and the supporting No Significant Hazards Determination

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose changes to the Facility Operating Licenses and Appendix A, Technical Specifications, as follows and as presented in the accompanying Parts B through G of this Package.

BACKGROUND

Over the past several years the nuclear industry and the Nuclear Regulatory Commission (NRC) have jointly developed Improved Standard Technical Specifications (ISTS). The NRC has encouraged licensees to implement these improved technical specifications as a means for improving plant safety through the more operator-oriented technical specifications, improved and expanded bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

This License Amendment Request (LAR) is submitted to conform the Prairie Island Nuclear Generating Plant (PINGP) Current Technical Specifications (CTS) to NUREG-1431, Improved Standard Technical Specifications, Westinghouse plants, Revision 1 issued April 1995 (ISTS). The resulting new Technical Specifications (TS) for Prairie Island (PI) are the PI Improved Technical Specifications (ITS) which incorporates the PI plant specific information.

NUREG-1431 is based on a hypothetical four loop Westinghouse plant. Since PI is similar in design and vintage to the R.E. Ginna Nuclear Power Plant which has already completed conversion to improved technical specifications, this amendment request relies on the Ginna ITS.

This LAR is also supported by Parts B through G. Part B contains a "clean" copy of the proposed PI ITS and Bases. Part C contains a mark-up of the PI CTS. Part D is the Description of Changes (DOC) to the PI CTS. Part E is a mark-up of the ISTS and Bases which shows the deviations from the standard incorporated to meet PI plant specific requirements. Part F gives the Justification for Deviations (JFD) from the ISTS and Part G provides the No Significant Hazards Determinations (NSHD) for changes to the PI CTS. To facilitate review of this LAR, cross-reference numbers from changes and deviations to the corresponding DOC, JFD and NSHD are provided. The methodology for mark-up and cross-references are described in the next section.

MARK-UP METHODOLOGY

The TS conversion package includes mark-ups of the CTS, the ISTS and the ISTS Bases in accordance with this guidance. Mark-up may be electronic or by hand as indicated.

Current Technical Specifications

The mark-up of the CTS is provided to show where current requirements are placed in the ITS, to show the major changes resulting from the conversion process, and to allow reviewers to evaluate significant differences between the CTS and ITS.

This ITS conversion LAR has been prepared in 14 packages following the Chapter/Section outline of the ITS as follows: 1.0, 2.0, 3.0, 3.1 . . . 3.9, 4.0 and 5.0. Accordingly, each package contains all the elements of Parts A through G as described above. The CTS Bases are not included in the CTS mark-up packages since the Bases have been rewritten in their entirety.

The current Specifications addressed by the associated ITS Chapter/Section are cross-referenced in the left margin to the new ITS location by Specification number and type (G-General, SL-Safety Limit, LCO-Limiting Condition for Operation or SR-Surveillance Requirements). Those portions of each CTS page which are not addressed in the associated ITS Chapter/Section are shadowed (electronic) or clouded and crossed out (by hand) and in the right margin is the comment, "Addressed Elsewhere".

The CTS are marked-up to incorporate the substance of NUREG-1431 Revision 1. It is not the intent to mark every nuance required to make the format change from CTS to ITS.

In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.

Some apparent changes result from the different conventions and philosophies used in the ITS. Generally these apparent changes will not be marked-up in the CTS if there is no resulting change in plant operating requirements.

Changes are identified by a change number in the right margin which map the changed specification requirement to Part D, Discussion of Changes, and Part G, No Significant Hazards Determination (NSHD) and indicate the NSHD category. The change number form is R3.4-02 where the first two numbers, 3.4 in this example, refer to ITS Chapter/Section number 3.4, and the second number, 02 in this example, is a sequentially assigned number for changes within that Chapter/Section, starting with 01. The prefix letter(s) indicates the classification of the change impact. For CTS changes this is also the NSHD category.

The change impact categories defined below conveniently group the type of changes for consideration of the effect of the change on the current plant license in Part D and are also useful for efficient discussion in Part G the "No Significant Hazards Determination" (NSHD) section. If the same change is made in Part E, then the change impact category will also show up in the change number in Part F. These categories are:

A - Administrative changes, editorial in nature that do not involve technical issues. These include reformatting, renaming (terminology changes), renumbering, and rewording of requirements.

L - Less restrictive requirements included in the PI ITS in order to conform to the guidance of NUREG-1431. Generally these are technical changes to existing TS which may include items such as extending Completion Times or reducing Surveillance Frequencies (extended time interval between surveillances). The less restrictive requirements necessitate individual justification. Each is provided with its specific NSHD.

LR - Less restrictive Removal of details and information from otherwise retained specifications which are removed from the CTS and placed in the Bases, Technical Requirements Manual (TRM), Updated Safety Analysis Report (USAR) or other licensee controlled documents. These changes include details of system design and function, procedural details or methods of conducting surveillances, or alarm or indication-only instrumentation.

M - More restrictive requirements included in the PI ITS in order to provide a complete set of Specifications conforming to the guidance of NUREG-1431. Changes in this category may be completely new requirements or they may be technical changes made to current requirements in the CTS.

R - Relocation of Current Specifications to other controlled documents or deletion of current Specifications which duplicate existing regulatory requirements.

Current requirements in the LCOs or SRs that do not meet the 10 CFR 50.36 selection criteria and may be relocated to the Bases, USAR, Core Operating Limits Report (COLR), Operational Quality Assurance Plan (OQAP), plant procedures or other licensee controlled documents. Relocating requirements to these licensee controlled documents does not eliminate the requirement, but rather, places them under more appropriate regulatory controls, such as 10CFR 50.54 (a)(3) and 10 CFR 50.59, to manage their implementation and future changes. Maintenance of these requirements in the TS commands resources which are not commensurate with their importance to safety and distract resources from more important requirements. Relocation of these items will enable more efficient maintenance of requirements under existing regulations and reduce the need to request TS changes for issues which do not affect public safety.

Deletion of Specifications which duplicate regulations eliminates the need to change Technical Specifications when changes in regulations occur. By law, licensees shall meet applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions; therefore, restatement in the Technical Specifications is unnecessary.

The methodology for marking-up these changes is as follows:

As discussed above, administrative changes may not be marked-up in detail. Portions of the specifications which are no longer included are identified by use of the electronic strike-out feature (or crossed out by hand). Information being added is inserted into the specification in the appropriate location and is identified by use of shading features (or handwritten/insert pages).

Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS mark-up is to identify changes from the ISTS required to create a plant specific ITS by incorporating plant specific values in bracketed fields and identifying other changes with cross-reference to the Part F Justification For Differences.

All deviations from the ISTS are cross-referenced to the Part F justification for differences by a change number in the right margin. The change number form is CL3.4-05 where the prefix letter(s), CL in this example, indicate the classification of the reason for the difference, the first two numbers, 3.4 in this example, refer to the ITS Chapter/Section number 3.4, and the second number, 05 in this example, is a sequentially assigned number for deviations within that Chapter/Section, starting with a number which is larger than the last number from the Part C CTS mark-up. In some instances where a change has been made to the CTS and ISTS, the Part D change number is given since the justification for difference is the same as the discussion of change. The following categories are used as prefixes to indicate the general reason for each difference:

CL - Current Licensing basis. Issues that have been previously licensed for PI and have been retained in the ITS. This includes Specifications dictated by plant design features or the design basis. Since no plant modifications have been or will be made to accommodate conversion to ITS, the plant design basis features shall be incorporated into the PI ITS.

PA - Plant, Administrative. Plant specific wording preference or minor editorial improvements made to facilitate operator understanding.

TA - Traveler, Approved. Deviations made to incorporate an industry traveler which has been approved by the NRC.

TP - Traveler, Proposed. Deviation made to incorporate a proposed industry traveler which as of the time of submittal has not been approved by the NRC.

X - Other, Deviation from the ISTS for any other reason than those given above.

Material which is deleted from the ISTS is identified by use of the WordPerfect strike-out feature (or crossed out by hand). Information being added to the ISTS to generate the PI ITS due to any of the deviations discussed above is identified by use of WordPerfect red-line features (or handwritten/insert pages).

Bracketed Information

Many parameters, conditions, notes, surveillances, and portions of sections are bracketed in the ISTS recognizing that plant specific values are likely to vary from the "generic" values provided in the standard.

If the bracketed value applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. In some instances, bracketed material is not discussed. If bracketed material is discussed, a change number is provided which includes the appropriate prefix as described above. When bracketed "generic" material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand), the plant specific information is substituted for the bracketed information and a change number is provided which includes the appropriate prefix. Information added is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages).

Optional Sections

Due to differing Westinghouse plant designs and methodologies, some ISTS section numbers include a letter suffix indicating that only one of these sections is applicable to any specific plant. The appropriate section is indicated in the Table of Contents, the suffix letter is deleted, and justification, if required, is included in the appropriate Chapter/Section package.

Bases, Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS Bases have been marked-up to support the plant specific PI ITS and allow reviewers to identify changes from NUREG-1431. To the extent possible, the words of NUREG-1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading with respect to Prairie Island, they have been revised. In addition, descriptions have been added to cover plant specific portions of the specifications. Change numbers have been provided for the ISTS Bases with the same format as the ISTS Specification mark-up. In some instances, the same change number is used to describe the change.

Material which is deleted from the ISTS Bases is identified by use of the strike-out feature of WordPerfect (or crossed out by hand). Information being added to the ISTS Bases to generate the PI ITS is identified by use of the red-line (shading) feature of WordPerfect (or handwritten/insert pages).

Bracketed Material

Many parameters and portions of Bases are bracketed in the ISTS recognizing that plant specific values and discussions are likely to vary from the "generic" information provided in the standard.

If the bracketed information applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. No change number or justification is provided for use of bracketed material, unless special circumstances warrant discussion.

When bracketed "generic" Bases material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand) and the plant specific information substituted for the bracketed information is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages). A change number with the same format as those used for the ISTS Specification mark-up is provided.

ACRONYMS

Many acronyms are used throughout this submittal. The intent of the final ITS (Part B) is that in general acronyms be written in full prior to the first use. Commonly used acronyms may not be written in full. Other parts of this package may not always write in full each acronym prior to first use; therefore, a list of acronyms is attached to assist in the review of this package.

Attachment to Part A

LIST OF ACRONYMS

AB	Auxiliary Building
ABSVS	Auxiliary Building Special Ventilation System
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ALARA	As Low As Reasonably Achievable
ALT	Actuation Logic Test
ASA	Applicable Safety Analyses
ASME	American Society of Mechanical Engineers
AOO	Anticipated Operational Occurrences
AOT	Allowed Outage Time
BAST	Boric Acid Storage Tank
BIT	Boron Injection Tank
BOC	Beginning of Cycle
CC	Component Cooling
COT	CHANNEL OPERATIONAL TEST
CAOC	Constant Axial Offset Control
CET	Core Exit Thermocouple
CL	Cooling Water
CLB	Current Licensing Basis
COLR	Core Operating Limits Reports
CRDM	Control Rod Drive Mechanism
CRSVS	Control Room Special Ventilation System
CS	Containment Spray
CST	Condensate Storage Tanks
CTS	Current Technical Specification(s)
DBA	Design Basis Accident
DDCL	Diesel Driven Cooling Water
DG	Diesel Generator
DNB	Departure from Nucleate Boiling
DNBR	Departure from nucleate boiling ratio
ECCS	Emergency Core Cooling System

EDG	Emergency Diesel Generators
EFPD	Effective Full Power Days
EOC	End of Cycle
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FWLB	Feedwater Line Break
GDC	General Design Criteria
GITS	Ginna Improved Technical Specifications
HELB	High Energy Line Break
HZP	Hot Zero Power
IPE	Individual Plant Evaluation
ISTS	Improved Standard Technical Specifications
ITC	Isothermal Temperature Coefficient
ITS	Improved Technical Specifications
LA	License Amendment
LAR	License Amendment Request
LBLOCA	Large Break LOCA
LCO	Limiting Conditions for Operation
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Overpressure Protection
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MFW	Main Feedwater
MOSCA	MODE or Other Specified Condition of Applicability
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
MSLB	Main Steam Line Break
MSLI	Main Steam Line Isolation
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
NMC	Nuclear Management Company
NPSH	Net Positive Suction Head

NRCV	Non-Return Check Valve
NUREG-1431	The ISTS for Westinghouse plants
OPPS	OverPressure Protection System
PCT	Peak Cladding Temperature
PI	Prairie Island
PITS	Prairie Island Technical Specifications
PIV	Pressure Isolation Valve
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valve
PTLR	Pressure and Temperature Limits Report
QTPR	Quadrant Power Tilt Ratio
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RPI	Rod Position Indication
RPS	Reactor Protection System
RTB	Reactor Trip Breaker
RTBB	Reactor Trip Bypass Breaker
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SBVS	Shield Building Ventilation System
SCWS	Safeguards Chilled Water System
SDM	Shut Down Margin
SFDP	Safety Function Determination Program
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SL	Safety Limit

SLB	Steam Line Break
SR	Surveillance Requirements
SSC	Structures, Systems and Components
TADOT	Trip Actuating Device Operational Test
TDAFW	Turbine Driven Auxiliary Feedwater
TRM	Technical Requirements Manual
TS	Technical Specifications
TSSC	Technical Specification Selection Criteria
TSTF	Term used for a NUREG change (traveler)
VCT	Volume Control Tank
VFTP	Ventilation Filter Test Program
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
WCAP	Westinghouse technical report

PACKAGE 3.9

REFUELING OPERATIONS

PART B

PROPOSED PRAIRIE ISLAND IMPROVED TECHNICAL SPECIFICATIONS AND BASES

List of Pages

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3.9.1-2	B 3.9.1-1	B 3.9.3-5	B 3.9.5-5
3.9.2-1	B 3.9.1-2	B 3.9.4-1	B 3.9.6-1
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3.9.6-1	B 3.9.3-2	B 3.9.5-2	
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

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3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

----------NOTE----------
Only applicable to the refueling cavity when connected to the RCS.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.9 REFUELING OPERATIONS

3.9.2 Refueling Cavity Water Level

LCO 3.9.2 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: 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 movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two neutron flux monitors and one neutron flux monitors audible countrate circuit shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. Two required neutron flux monitors inoperable.	B.1 Initiate action to restore one neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Neutron flux monitor audible countrate circuit inoperable.	C.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	C.3 Perform SR 3.9.1.1	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ----------	
	Perform CHANNEL CALIBRATION.	24 months

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed, or capable of being closed with at least two containment fan coil unit fans capable of operating in high speed; and

c. Each penetration providing direct access from the containment atmosphere to the environment either:

1. closed by a manual or automatic isolation valve, or blind flange; or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	----------NOTE---------- Not required to be met for containment inservice (low flow) purge valve(s) in penetrations closed to comply with LCO 3.9.4.c.1. ---------- Verify each required containment inservice (low flow) purge system valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation-High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

----------NOTE----------
The required RHR loop may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System, coolant with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 20 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	
	A.4 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	A.5 Close one door in each airlock.	4 hours
	AND	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, or blind flange.	4 hours
	OR	
	A.6.2 Verify each inservice (low flow) purge penetration is capable of being closed by an OPERABLE Containment Ventilation Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.5.1 Verify one RHR loop is in operation.	12 hours

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

----------NOTES----------

1. The required operating RHR loop may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction onto the Reactor Coolant System (RCS), coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

2. The required operating RHR loop may be removed from operation, and considered OPERABLE and in operation, to support filling and draining the reactor cavity when aligned to, or during transitioning to or from, the refueling water storage tank provided the required RHR loop is capable of being realigned to the RCS.

APPLICABILITY: MODE 6 with the water level < 20 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loop(s) to OPERABLE status. OR	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Initiate action to establish ≥ 20 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	AND	
	B.3 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	B.4 Close one door in each air lock.	4 hours
	AND	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, or blind flange.	4 hours
	OR	
	B.5.2 Verify each penetration is capable of being closed by an OPERABLE Containment Ventilation Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation.	12 hours
SR 3.9.6.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration and associated shutdown margin limits are specified in the COLR. The required boron concentration will vary depending on time in core life. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

AEC GDC Criterion 27 (Ref. 1) requires that two independent reactivity control systems of different design principles be provided. AEC GDC Criterion 29 (Ref. 1) requires at least one of these systems must be capable of holding the reactor core subcritical under any condition. The Chemical and Volume Control System (CVCS), Safety Injection (SI) and Residual Heat Removal (RHR) are the systems capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling cavity is then flooded with borated water from the refueling water storage

BASES

BACKGROUND (continued)

tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling cavity above the appropriate COLR limits.

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 for MODE 6. The boron concentration limits specified in the COLR are based on the core reactivity at one or more points in the fuel cycle and include 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, an adequate margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer 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 analyzed boron dilution accident requiring the highest boron concentration occurs in MODE 6 (Ref. 2).

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

BASES (continued)

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling cavity while in MODE 6. The boron concentration limits specified in the COLR ensure that a core k_{eff} of ≤ 0.95 or other lower value based on the dilution analysis 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} ≤ 0.95 or a lower appropriate level based on the dilution analysis. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling cavity when connected to the RCS. When the refueling cavity is isolated from the RCS, no potential path for boron dilution exists.

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 concentration of any coolant volume in the RCS or the refueling cavity, when connected, is less than that needed to maintain shutdown margin within 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.

BASES

ACTIONS

A.1 and A.2 (continued)

Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.3

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, and connected portions of the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling cavity to the RCS, this SR must

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1 (continued)

be met per SR 3.0.4. If any dilution activity has occurred while the cavity was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

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. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27 and 29, issued for comment July 10, 1967, as referenced in USAR, Section 1.

2. USAR, Chapter 14.4.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel 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, 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 < 25% of 10 CFR 100 limits.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Reference 1. A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (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 100 hours prior to fuel handling, 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 (Refs. 3 and 4).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

BASES (continued)

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.

APPLICABILITY

LCO 3.9.2 is applicable 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. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.15, "Spent Fuel Storage Pool Water Level."

ACTIONS

A.1

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

The suspension of fuel movement shall not preclude completion of movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1 (continued)

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.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. USAR, Section 14.5.

3. 10 CFR 100.10.

4. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

Neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors (N-31, N-32, N-51, and N-52) are located external to the reactor vessel and detect neutrons leaking from the core.

The installed neutron flux monitors are:

a. BF3 detectors operating in the proportional region of the gas filled detector characteristic curve; or

b. Fission chambers.

The detectors monitor the neutron flux in counts per second. The instrument range used for monitoring changes in subcritical multiplication typically covers six decades of neutron flux. The detectors provide continuous visual indication in the control room. The installed BF3 neutron flux monitors provide an audible indication to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

Two OPERABLE neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly.

The neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

BASES (continued)

LCO

This LCO requires that two neutron flux monitors, capable of monitoring subcritical neutron flux, be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. Neutron detectors N-31, N-32, N-51 and N-52 may be used to satisfy this LCO requirement.

This LCO also requires that one audible countrate circuit, associated with either N-31 or N-32, be OPERABLE to ensure that audible indication is available to alert the operator in the event of a dilution accident.

APPLICABILITY

In MODE 6, the 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 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 required neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

BASES

ACTIONS

A.1 and A.2 (continued)

Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not result in reducing SDM below the required value. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

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

B.2

With no required neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions that could lead to reducing SDM below the required value are not to be made, the core reactivity condition is stabilized until the 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

ACTIONS (continued)

C.1, C.2 and C.3

With no audible neutron flux monitor countrate circuit OPERABLE, only visual indication is available and prompt and definite indication of a boron dilution event would be lost. Performance of Required Actions C.1 and C.2 shall not preclude completion of movement of a component to a safe position or that is a normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity is stabilized until the audible count rate capability is restored. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of 12 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensure 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.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 13, 19, 27 and 31, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 14.4.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

BASES

BACKGROUND (continued)

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.

During movement of irradiated fuel assemblies within containment, containment closure or closure capability is required; therefore, the door interlock mechanism may remain disabled and both doors may be open provided one door can be closed with at least two containment fan coil unit fans capable of operating in high speed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will restrict fission product radioactivity release from containment to be within regulatory limits.

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a minipurge system referred to as containment inservice purge, includes a 14 inch purge penetration and an 18 inch exhaust penetration.

During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position, or the penetrations may be blank flanged. The two valves in each of the two containment inservice purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation System.

BASES

BACKGROUND (continued)

In MODE 6, sufficient air flow rates are necessary to conduct refueling operations. The inservice purge system is used for this purpose, and each of the four valves is closed by the radiation monitors associated with the containment inservice purge system in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation." The 36 inch subsystem is normally blank flanged, although the option for use is allowed during outages, except during movement of irradiated fuel or CORE ALTERATIONS.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, or blind flange.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.2, "Refueling Cavity Water Level," in conjunction with the minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. The acceptance limit for offsite radiation exposure is 25% of 10 CFR 100 values.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will restrict fission product release from containment to be well within regulatory limits. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A fuel handling accident does not cause containment pressurization; however, with an assumed single failure, the operating purge system supply fan is assumed to continue supplying air to containment. To maintain post-fuel handling accident releases well within the limits of 10 CFR 100, only the inservice purge system is allowed to be operating during fuel movement. Two fan coil unit fans are required to operate in the high speed mode following a fuel handling accident to assure that radioactive material in containment is well mixed and any releases will leave containment at a lower concentration over the duration of the accident. The provision that one air lock door is OPERABLE and under procedural control will assure that at least one door will be closed within 30 minutes as required, thus assuring radioactive releases are well within the limits of 10 CFR 100 (Ref. 1).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment.

The LCO requires containment penetrations to meet the following requirements:

a. The equipment hatch is closed and held in place by at least 4 bolts;

b. One door in each air lock is closed, or capable of being closed with at least two containment fan coil unit fans capable of operating in high speed; and

BASES

LCO (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

1. closed by a manual or automatic isolation valve, blind flange, or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

The containment air lock doors may be open during movement of irradiated fuel in the containment provided that one door is capable of being closed within 30 minutes in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one door in each air lock will be closed following an evacuation of containment.

For the OPERABLE containment inservice purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. The OPERABILITY requirements for this LCO require that the automatic containment inservice purge and exhaust valve closure times specified in the IST Program can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during movement of 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.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the inservice 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 irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open inservice purge and exhaust valves will demonstrate that the valves will function if required during a fuel handling accident. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic Containment Ventilation Isolation signal.

The Surveillance is performed every 7 days during movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance is to be conducted before the start of refueling operations and then in

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

accordance with the frequency specified. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment inservice purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances, when performed, will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. USAR, Section 14.5.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation-High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), to provide mixing of borated coolant, and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel which would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 20 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

The RHR System, during refueling conditions, satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 20 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability.

At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes a RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to a RCS cold leg.

The LCO is modified by a Note that allows the required operating RHR loop to not be in operation for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction, with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained, is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

BASES (continued)

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 20 ft above the top of the reactor vessel flange, to provide decay heat removal.

Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 20 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations.

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not result in reducing core reactivity below the required SDM or refueling boron concentration limit.

BASES

ACTIONS (continued)

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 20 ft above the reactor vessel flange provides an adequate available heat sink.

Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

In accordance with LCO 3.9.2, "Refueling Cavity Water Level," movement of irradiated fuel within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 20 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4, A.5, A.6.1, and A.6.2

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts;

BASES

ACTIONS

A.4, A.5, A.6.1, and A.6.2 (continued)

b. One door in each air lock must be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, or blind flange, or verified to be capable of being closed by an OPERABLE Containment Ventilation Isolation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil, clad to fail, and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows adequate time to fulfill the Required Actions and not exceed dose limits.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation in order to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

None.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level

BASES

BACKGROUND	The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), to provide mixing of borated coolant, and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg. Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.
APPLICABLE SAFETY ANALYSES	If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel, which could eventually challenge the integrity of the fuel cladding, a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge. The RHR System, during refueling conditions, satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).
LCO	In MODE 6, with the water level < 20 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

BASES

LCO (continued)

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold leg.

Either RHR pump may be aligned to the Refueling Water Storage Tank (RWST) to support filling or draining the refueling cavity or for performance of required testing.

The LCO contains two Notes which provide clarification of the LCO.

Note 1 allows the required operating RHR loop to not be in service for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration with coolant at concentrations less than required to meet the minimum boron concentration of LCO 3.9.1. The RHR loop only needs to be running if needed.

Note 2 allows a required RHR loop to be removed from operation, but considered OPERABLE and in operation during filling or draining of the refueling cavity, when aligned to or during transitioning to or from the RWST, provided the required RHR loop is capable of being realigned to the RCS. This variance allows the unit to perform necessary evolutions and recognizes that the RHR loop is still functional and that the system can be realigned easily. This note allows the full flow test of RHR to the upper head injection path.

BASES (continued)

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 20 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 20 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loop(s) are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 20 ft of water level is established above the reactor vessel flange. When the water level is ≥ 20 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes,

BASES

ACTIONS

B.1 (continued)

including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not result in reducing core reactivity below the required SDM or refueling boron concentration limit.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B5.1 and B.5.2

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts;

b. One door in each air lock must be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, or blind flange, or verified to be capable of being closed by an OPERABLE Containment Ventilation Isolation System.

With the RHR loop requirements not met, the potential exists for the coolant to boil, clad to fail, and release radioactive gas to the containment atmosphere. Performing the actions described above

BASES

ACTIONS

B.3, B.4, B5.1 and B.5.2 (continued)

ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows adequate time to fulfill the Required Actions and not exceed dose limits.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core.

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional RHR pump can be placed in operation, if needed, to maintain decay heat removal and circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

PACKAGE 3.9

REFUELING OPERATIONS

PART C

MARKUP OF PRAIRIE ISLAND CURRENT TECHNICAL SPECIFICATIONS

List of Pages

Part C Page	Current Technical Specifications Page
1	Table TS.1-1
2	TS.3.8-1
3	TS.3.8-1 (Overflow)
4	TS.3.8-2
5	TS.3.8-2 (Overflow)
6	TS.3.8-2 (Overflow) (continued)
7	Table TS.4.1-2A (Page 1 of 2)
8	Table TS.4.1-2A (Page 1 of 2)(Overflow)
9	Table TS.4.1-2B (Page 1 of 2)
10	Table TS.4.1-2B (Page 2 of 2)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications Conversion Submittal

TABLE TS.1-1

Addressed Elsewhere

OPERATIONAL MODES

MODE	TITLE	REACTIVITY CONDITION	%RATED THERMAL POWER	AVERAGE COOLANT TEMPERATURE	REACTOR VESSEL HEAD CLOSURE BOLTS FULLY TENSIONED
1	POWER OPERATION	Critical	> 2%	NA	YES
2	HOT STANDBY**	Critical	≤ 2%	NA	YES
3	HOT SHUTDOWN**	Subcritical	NA	≥ 350°F	YES
4	INTERMEDIATE SHUTDOWN**	Subcritical	NA	< 350°F ≥ 200°F	YES
5	COLD SHUTDOWN	Subcritical	NA	< 200°F	YES
6	REFUELING	NA*	NA	NA	NO

A3.9-05

* Boron concentration of the reactor coolant system and the refueling cavity (when connected) sufficient to ensure that

LCO3.9.1

~~the more restrictive of the following conditions is met:~~

LR3.9-01

~~a. $K_{eff} \leq 0.95$, or~~

~~b. Boron concentration ≥ 2000 ppm.~~

c. Shutdown Margin limits are met as specified in the Core Operating Limits Report.

LCO3.9.1 Cond A

New Required Actions are provided which specify the actions when the boron concentration limits are not met.

M3.9-08

~~** Prairie Island specific MODE title, not consistent with Standard Technical Specification MODE titles. MODE numbers are consistent with Standard Technical Specification MODE numbers.~~

Addressed Elsewhere

~~3.8 REFUELING AND FUEL HANDLING~~

~~Applicability~~

~~Applies to operating limitations associated with fuel-handling operations and CORE ALTERATIONS.~~

~~Objectives~~ A3.9-02

~~To ensure that no incident could occur during fuel handling and CORE ALTERATIONS that would affect public health and safety.~~

Specification

A. ~~Core Alterations~~ A3.9-03

1. ~~During CORE ALTERATIONS~~ the following conditions shall be satisfied (except as specified in 3.8.A.2 and 3 below):

LCO 3.9.4 a. During movement of irradiated fuel assemblies in containment L3.9-04

1) The equipment hatch shall be closed and held in place by four bolts. In addition, at least one isolation valve shall be ~~OPERABLE or locked~~ closed by a manual or automatic valve, or blind flange, or A3.9-06 A3.9-07

in each line which penetrates the containment and provides a direct path from containment atmosphere to the outside atmosphere, ~~.~~ OR A3.9-10

2) Airlock doors

a) At least one door in each air lock is closed, or capable of being closed and A3.9-10

b) ~~Both doors in each air lock may be open if:~~ A3.9-10

~~i. The containment (high flow) purge system is isolated,~~ LR3.9-11

ii. ~~The inservice (low flow) purge system is~~ A3.9-12 capable of automatic isolation by the Containment Ventilation Isolation System, LR3.9-13

iii. ~~At least one door in each air lock is OPERABLE, under procedural control, and capable of being closed within 30 minutes following a fuel handling accident in containment,~~ and

iv. At least two containment fan coil unit fans are capable of operating in the high speed mode ~~following a fuel handling accident in containment~~. A3.9-14

~~TS 3.8-1~~
~~Overflow~~

SR3.9.4.1
SR3.9.4.2

New SRs. Verify containment penetration status, verify inservice (low flow) purge system valves actuate to isolation position on actual or simulated signal (not applicable if the valves are closed).

M3.9-16

~~b. Radiation levels in the fuel handling areas of the containment shall be monitored continuously.~~

LR3.9-17

~~3.8.A.1.c.~~ The core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, ~~each with continuous visual~~ LCO3.9.3 LR3.9-18

~~indication in the control room~~ and one with audible indication ~~in the containment~~, which are in service whenever in MODE 6 M3.9-21

LCO3.9.3 Cond A ~~core geometry is being changed~~. When ~~core geometry is not being changed, at least~~ only one required neutron flux monitor is ~~shall be~~ in service take 3.8.A.2 required action.

LCO3.9.3 Cond B When both required neutron flux monitors are inoperable, verify boron concentration every 12 hours. M3.9-22

LCO3.9.3 Cond C If the audible countrate indication is not operable, verify boron concentration every 12 hours. M3.9-23

SR3.9.3.1 SR3.9.3.2 New SRs, perform CHANNEL CHECK and CHANNEL CALIBRATION. M3.9-24

~~d. The plant shall be in the REFUELING condition.~~ A3.9-26

LR3.9-27

e. During movement of fuel assemblies in containment ~~or control rods out of the reactor vessel~~, at least 23 feet of water shall be maintained above the reactor vessel flange. The required water level shall be verified ~~prior to moving fuel assemblies or control rods and~~ at least once every day ~~while the cavity is flooded~~. LCO3.9.2 A3.9-28 SR3.9.2.1 L3.9-31

f. At least one residual heat removal pump shall be OPERABLE and running in MODE 6 with water level > 20 ft above the top of reactor vessel flange and in MODE 6 with the water level < 20 ft above the top of reactor vessel flange. LCO3.9.5 LCO3.9.6 A3.9-32

LCO3.9.5 Note LCO3.9.6 Note 1 The pump may be shut down for up to one hour per 8 hour period provided no operations are permitted which would cause reduction of RCS boron concentration. ~~to facilitate movement of fuel or core components.~~ M3.9-33 L3.9-34

SR3.9.5.1 New SR, verify RHR loop is in operation. M3.9-36

3.8.A.1.g (continued)

~~g.~~ In Mode 6, If the water level above the top of the reactor vessel flange is less than 20 feet, — M3.9-37

LCO3.9.6

~~except for control rod unlatching/latching operations or upper internals removal/replacement,~~ — M3.9-37

both residual heat removal loops shall be OPERABLE.

LCO3.9.6 Note 2 — New LCO NOTE 2 which allows the operating RHR to be removed from operation and considered OPERABLE and in operation. — M3.9-37

SR3.9.6.1 SR3.9.6.2 — New SRs, verify one RHR loop in operation, verify breaker alignment and power available to other RHR pump. — M3.9-43

~~h. Direct communication between the control room and the operating floor of the containment shall be available whenever CORE ALTERATIONS are taking place.~~ — LR3.9-44

~~i. No movement of irradiated fuel in the reactor shall be made untilthe reactor has been subcritical for at least 100 hours.~~ — LR3.9-46

Addressed Elsewhere

j. The radiation monitors which initiate isolation of the Containment Purge System shall be tested and verified to be OPERABLE prior to CORE ALTERATIONS.

LCO3.9.2 Cond A
LCO3.9.3 Cond A&C
LCO3.9.4 Cond A

2. If ~~any of~~ the above conditions are not met, CORE ALTERATIONS shall cease (for 3.8.A.1.c (ITS 3.9.3)), or cease irradiated fuel movement in containment (for 3.8.A.1.a (ITS 3.9.4) and e (ITS 3.9.2)). — A3.9-47

LCO3.9.3 Cond B
LCO3.9.5 Cond A
LCO3.9.6 Cond A

A3.9-47

Work shall be initiated to correct the violated conditions so that the specifications are met (for 3.8.A.1.c (ITS 3.9.3), f (ITS 3.9.5) and g (ITS 3.9.6)).

LCO3.9.3 Cond A
LCO3.9.5 Cond A
LCO3.9.6 Cond B

A3.9-47

and (for 3.8.A.1.c (ITS 3.9.3), f (ITS 3.9.5) and g (ITS 3.9.6)) no operations which may increase the reactivity of the core shall be performed

~~TS 3.8-3~~
~~Overflow~~
~~continued~~

~~3~~. If Specification 3.8.A.1.f or 3.8.A.1.g cannot be satisfied, all fuel handling operations in the core ~~containment~~ shall be A3.9-47

LCO3.9.5 Cond A

suspended (3.8.A.1.f (ITS 3.9.5)), the requirements of A3.9-47

LCO3.9.5 Cond A
LCO3.9.6 Cond B

Specification 3.8.A.1.a.1) shall be satisfied,

at least one door in each personnel air lock shall be closed,

LCO3.9.5 Cond A
LCO3.9.6 Cond B

and no reduction in reactor coolant boron concentration less than required to meet LCO 3.9.1 shall be made. A3.9-48

~~Table TS 4.1-2A (Page 1 of 2)~~

Addressed Elsewhere

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

Equipment	Test	Frequency	~~FSAR Sect. Reference~~
1. Control Rod Assemblies	Rod Drop Times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods	7
2. Control Rod Assemblies	Partial movement of all rods	Every Quarter	7
3. Pressurizer Safety Valves	Verify OPERABLE in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	
4. Main Steam Safety Valves	Verify each required lift setpoint in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	
5. Reactor Cavity	Water Level	~~Prior to moving fuel assemblies or control rods and~~ at least once every day during movement of irradiated fuel within containment ~~while the cavity is flooded.~~	

SR3.9.2.1

L3.9-31

~~TABLE TS.4.1-2A (Page 1 of 2) REV 123 5/21/96~~

~~Table TS.4.1-2A (Page 1 of 2) (Overflow)~~

Addressed Elsewhere

6.	Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1),(b).2 or 3.1.A.2.c.(1),(b).3
7.	Pressurizer PORVs	Functional	Every 18 months

~~TABLE TS.4.1-2A (Page 1 of 2) Overflow~~

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

Addressed Elsewhere

TEST	FREQUENCY
1. RCS Gross Activity Determination	5/week
2. RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3. RCS Radiochemistry $\bar{E}$ determination	1/6 months (1) (when at power)
4. RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5. RCS Radiochemistry (2)	Monthly
6. RCS Tritium Activity	Weekly
7. Deleted	

SR3.9.1.1

8. RCS Boron Concentration*(3) 2/Week (4)

Addressed Elsewhere

9. RWST Boron Concentration	Weekly
10. Boric Acid Tanks Boron Concentration	2/Week
11. Caustic Standpipe NaOH Concentration	Monthly
12. Accumulator Boron Concentration	Monthly
13. Spent Fuel Pit Boron Concentration	Weekly

* Required at all times

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

Addressed Elsewhere

TEST	FREQUENCY
14. Secondary Coolant Gross Beta-Gamma activity	Weekly
15. Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)
16. Secondary Coolant Chemistry	
pH	5/week (6)
pH Control Additive	5/week (6)
Sodium	5/week (6)

Notes:

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

2. To determine activity of corrosion products having a half life greater than 30 minutes.

3. During REFUELING, the boron concentration shall be verified by chemical analysis every three days ~~daily~~.

SR3.9.1.1

L3.9-49

4. The maximum interval between analyses shall not exceed 5 days.

5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.

6. The maximum interval between analyses shall not exceed 3 days.

Addressed Elsewhere

PACKAGE 3.9

REFUELING OPERATIONS

PART D

DISCUSSION OF CHANGES
(DOC)

to

PRAIRIE ISLAND
CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Part D

Package 3.9

REFUELING OPERATIONS

DISCUSSION OF CHANGES TO CURRENT TECHNICAL SPECIFICATIONS

The proposed changes to PI Operating License Appendix A, TS are discussed below and the specific wording changes are shown in parts B, C and E.

For ease of review, all package part and discussions are organized according to the proposed PI ITS Table of Contents.

NSHD Category	Change Number 3.9-	Discussion of Change
LR	01	Table 1-1, Definition of Operational Modes, Footnote * The requirement to maintain the refueling boron concentration at a level which assures $K_{eff} \leq 0.95$ and boron concentration ≥ 2000 will be relocated to the COLR. The COLR will specify the SDM required for all analyzed plant refueling conditions which assures that $K_{eff} \leq 0.95$ and boron concentration is ≥ 2000 ppm. Any changes to values in the COLR are made using NRC approved methodologies. Therefore it is unnecessary to retain these requirements in the TS. This change is generally consistent with the guidance of NUREG-1431 which allows the boron concentrations to be specified in the COLR. Since the COLR is licensee controlled, this change is less restrictive.

NSHD Category	Change Number 3.9-	Discussion of Change
A	02	CTS 3.8 Applicability and Objectives. The beginning of each CTS section contains general statements of Applicability and Objectives for that TS section. This Applicability states the plant operations to which the specifications apply which is a different meaning than the Applicability in NUREG-1431. Since the ITS clearly states within each specification the operations to which it applies, administratively these statements have been incorporated. Likewise, the CTS Objectives statement provides an overall purpose for the specifications within the section. These objectives are administratively incorporated in general through the statement of the ITS specification LCO and the supporting Bases. Since these general CTS statements do not establish any regulatory requirements and are incorporated in a broad sense in the ITS, these are considered administrative changes.
A	03	3.8.A.1. Although the title for TS.3.8.A is "Core Alterations" many of the individual specifications in this section state other applicabilities. Likewise, the lead-in paragraph for 3.8.A.1 states that these specifications are applicable during "CORE ALTERATIONS" even though many of the individual specifications state other applicabilities. In the PI ITS each specification in the Refueling Operations Section contains its own Applicability and the CTS specifications are marked up to correspond to these applicabilities. Therefore, this change is administrative in that the applicability is removed here and will be addressed with each separate specification as appropriate.

NSHD Category	Change Number 3.9-	Discussion of Change
L	04	3.8.A.1.a. This change specifies that the applicability for Containment Penetrations is during movement of irradiated fuel assemblies in containment. This change is consistent with a Traveler which will be discussed in Part F of this package. The accidents postulated to occur during core alteration, in addition to fuel handling accidents, are: inadvertent criticality (due to a control rod removal error or continuous rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with a fuel assembly in an improper location. These events have been analyzed at PI and do not result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, this specification is revised to only apply during fuel handling. Since this change will reduce the evolutions requiring control of containment penetrations, it is a less restrictive change.
A	05	CTS Table 1-1, Definition of Operational Modes, Footnote *. The definition of areas required to have the specified boron concentration includes the refueling cavity. This requirement has been modified by the clause, "when connected". This is a clarification since, the cavity does not have any impact on the core reactivity when it is not flooded or otherwise connected to the reactor coolant system. Since this does not substantively change any plant operations or testing requirements, this is an administrative change.

NSHD Category	Change Number 3.9-	Discussion of Change
A	06	3.8.A.1.a.1. CTS for containment closure require the equipment hatch to be closed but does not specify what it means to be "closed". This change defines four bolts required to hold the equipment hatch in place when it is considered closed. Since either more or less bolts could be used to hold the hatch in place in accordance with the CTS, this change is administrative. This is not a safety concern since during refueling, there is not a potential for containment pressurization and the hatch is not required to be leak tight in accordance with 10CFR50, Appendix J. This change is consistent with the guidance of NUREG-1431.
A	07	3.8.A.1.a.1. CTS requires at least one isolation valve to be OPERABLE (that is by automatic closure) or locked closed. For clarity, the CTS wording has been revised to be consistent with the wording in NUREG-1431. Since the meaning is the same this is an administrative change. This change is consistent with the guidance of NUREG-1431.
M	08	CTS Table 1-1, Definition of Operational Modes, Footnote *. New Required Actions are provided in the ITS which specify the actions to be taken when the RCS and refueling cavity (when connected) boron concentration is less than the required limit. These actions are acceptable since they are consistent with current plant practices and will not cause the plant to be operated in an unsafe manner. These new requirements have been included to make the ITS complete and consistent with the guidance of NUREG-1431. Since these are TS requirements, this is a more restrictive change.
	09	Not used.

NSHD Category	Change Number 3.9-	Discussion of Change
A	10	CTS 3.8.A.1.a.1), 2).a and 2).b). The CTS requirements have been rearranged to be consistent with NUREG-1431 and a Traveler which will also be discussed in Part F of this package. The intent of the Traveler is to allow containment airlock doors to be open during handling of fuel in containment. The option of keeping airlock doors open during fuel handling was previously approved for PI in License Amendment 119/112. (The License Amendment was actually broader in application in that it allows the airlock doors to be open during CORE ALTERATIONS.) The proposed wording in this specification differs from the Traveler to: 1) include the CTS requirement that two containment fan coil units are available to operate; and 2) retain the option of closing an airlock door if the other requirements are not met. For clarity, "atmosphere" has been added following "outside" to make this read the same as NUREG-1431. These changes are administrative since no substantive changes from CTS requirements are proposed.
LR	11	3.8.A.1.a.2).b).i. The CTS requirement to close the containment (high flow) purge system during fuel handling with the containment doors open is not included in the ITS. This requirement will be relocated to the TRM. This requirement will continue to be under the regulatory controls of 10CFR50.59 since the TRM is part of the USAR. Since the TRM is under licensee control, this is a less restrictive change. This change is acceptable since the containment high flow purge system is not used during fuel handling operations.

NSHD Category	Change Number 3.9-	Discussion of Change
A	12	3.8.A.1.a.2).b).ii. The CTS phrasing has been revised to remove "inservice (low flow) purge system" and include "Containment Ventilation Isolation System". These changes make the wording consistent with the intent of NUREG-1431 and since the Containment Ventilation Isolation System isolates the inservice (low flow) purge system, there is no change in meaning. Also to be consistent with NUREG-1431, this clause is relocated to be with the discussion of requirements for lines which provide a direct path to the environment. Since the intent of the specification is unchanged this is an administrative change.
LR	13	3.8.A.1.a.2).b).iii. CTS requirements for control of the airlock doors are relocated to the Bases, since this is unnecessary detail in the TS. This change is consistent with the guidance of NUREG-1431 and TSTF-51. Since the ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) are licensee controlled, this change is less restrictive.
A	14	3.8.A.1.a.2).b).iv. The clause, "following a fuel handling accident in containment" is not included in the ITS. It is understood that the provisions for closing an airlock door and operating the containment fan coil unit fans are to mitigate the consequences of a fuel handling accident in containment since that is the purpose of this specification. Therefore this clause is unnecessary. Since the intent of the specification is unchanged this is an administrative change.
	15	Not used.

NSHD Category	Change Number 3.9-	Discussion of Change
M	16	New SRs, 3.9.4.1 and 3.9.4.2 are included which require verification of containment penetration status and verification of inservice purge valve actuation. These are activities which are currently performed under plant procedures, therefore this change does not adversely impact plant operations. Since these will be formal TS required surveillances, this change is considered more restrictive. This change is included to make the PI ITS complete and consistent with the guidance of NUREG-1431.
LR	17	CTS 3.8.A.1.b. The CTS requirement for containment radiation monitors which provide monitoring for personnel safety, was not included in the PI ITS. No TS screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the containment radiation monitors are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the USAR (TRM). This is acceptable since the TRM is under the controls of 10CFR50.59. This change is consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.9-	Discussion of Change
LR	18	3.8.A.1.c. The CTS requirement for the neutron flux monitor to have continuous visual indication in the control room is relocated to the Bases. Visual indication is a normal part of considering these monitors operable and thus specification of this requirement in the TS is unnecessary detail. Likewise, the requirement to have audible indication in containment is relocated to the Bases. These changes are consistent with the guidance of NUREG-1431. Since the ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) are licensee controlled, these changes are less restrictive.
	19	Not used.
	20	Not used.
M	21	CTS 3.8.A.1.c. The applicability for this specification is revised to the broader requirement during MODE 6 which is consistent with the guidance of NUREG-1431. Neutron flux monitoring is required during MODE 6 since this is the only direct means of determining core reactivity changes during these activities. Since this change only involves more extensive use of monitoring equipment, it does not cause any conditions adverse to plant operations. This change is more restrictive since it requires more extensive monitoring.

NSHD Category	Change Number 3.9-	Discussion of Change
M	22	3.8.A.1.c. A new Action Statement is included which addresses inoperability of both required neutron monitors. This new action statement will require additional verification of boron concentration. This change is included to make the ITS complete and consistent with the guidance of NUREG-1431. Since this change requires additional verifications, it is more restrictive.
M	23	CTS 3.8.A.1.c. A new Action Statement is included which addresses inoperability of the audible countrate indication. This new action statement will require additional verification of boron concentration. This change is included to provide additional assurance that a boron dilution event will not occur. Since this change requires additional verifications, it is more restrictive.
M	24	New SRs, 3.9.3.1 and 3.9.3.2 are included which require CHANNEL CHECK and CHANNEL CALIBRATION of the neutron flux monitors. These are activities which are currently performed under plant procedures, therefore this change does not adversely impact plant operations. Since these will be formal TS required surveillances, this change is considered more restrictive. This change is included to make the PI ITS consistent with the guidance of NUREG-1431.
	25	Not used.

NSHD Category	Change Number 3.9-	Discussion of Change
A	26	3.8.A.1.d. Since each specification has its individual Applicability defined, this general applicability statement is unnecessary. This change is administrative since the applicability for each specification is addressed separately.
LR	27	CTS 3.8.A.1.e. CTS requires 23 feet of water above the reactor vessel flange during movement of control rods out of the reactor vessel. This change will relocate this requirement to the TRM. This change is consistent with the guidance of NUREG-1431. Since the TRM (under the regulatory control of 10 CFR 50.59) is licensee controlled, this change is less restrictive.
A	28	3.8.A.1.e. For consistency with the guidance of NUREG-1431, the applicability of this specification has been modified to apply to movement of fuel assemblies "in containment". This is an administrative change since it is understood in CTS that these water level requirements apply to activities in containment.
	29	Not used.
	30	Not used.

NSHD Category	Change Number 3.9-	Discussion of Change
L	31	CTS 3.8.A.1.e. The CTS requirement to verify water level prior to moving fuel or control rods is not included in the ITS. The requirement to verify water level while the cavity is flooded is not included since this specification has an applicability statement. The general rules of use and applicability for ITS (SR 3.0.4) require that all surveillance requirements be successfully met and current prior to entering the Mode of Applicability. Thus the water level will be verified prior to commencing movement of fuel. Thereafter, the SR requires verification of water level every 24 hours which is based on engineering judgement 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. Verification of water level when fuel movement is not occurring is not required because credit is not taken for this level of water in any accident analyses when this activity is not occurring. This change is consistent with the guidance of NUREG-1431.
A	32	3.8.A.1.f. An Applicability statement is added to this specification to clarify when this CTS paragraph is applicable. Since CTS paragraph 3.8.A.1.g specifies 20 ft as the level at which two RHR pumps are required to be operable, the 20 ft level is used in these applicability statements. The pool is full at 23.5 ft above the top of the reactor vessel flange. Since the plant does not have installed level indication at this elevation, it would be difficult for the operators to maintain the level within such a tight band thoughtout the outage. For this reason the CTS requires 20 and this level is retained in the ITS. To assure there are no ambiguity, a note is included that allows the RHR loop to be inoperable when the refueling cavity is being filled or drained. This note confirms current plant practices. Since these changes do not involve any substantive changes in plant operating requirements, these are administrative changes.

NSHD Category	Change Number 3.9-	Discussion of Change
M	33	3.8.A.1.f. CTS allow the operating RHR pump to be shut down for up to one hour to facilitate movement of fuel or core components. This change will prohibit operations which would cause reduction of RCS boron concentration. This change will not adversely affect safe plant operation and may improve plant safety. This change is included for consistency with NUREG-1431. Since additional limitations are placed on plant operations, this change is more restrictive.
L	34	3.8.A.1.f. CTS allow the operating RHR pump to be shut down for up to one hour to facilitate movement of fuel or core components. This change will remove the specific activities for which the RHR pump may be shut down. Since the specific activities for which the pump can be inoperable are not included in the TS this is a less restrictive change. This change is acceptable because there are adequate controls on plant activities when no RHR pumps are operating in Mode 6 and the time spent without an RHR pump operating is limited to one hour every eight hours. This change is included for consistency with NUREG-1431, Specification 3.9.5.
	35	Not used.
M	36	A new SR, 3.9.5.1, is included which requires verification that the RHR loop is in operation every 12 hours. Plant operators currently verify RHR pump operation, thus this change will not adversely affect plant operations. Since this verification will be a formal TS required surveillance, this change is considered more restrictive. This change is included to make the PI ITS complete and consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.9-	Discussion of Change
M	37	CTS 3.8.A.1.g. The CTS Applicability for this specification is revised to require applicability in all of Mode 6 to be consistent with the ITS. CTS 3.1.A.1.d requires two RHR loops to be operable when the water level is at or below the vessel flange and CTS 3.8 only applies during Core Alterations, thus this is a more restrictive requirement.. This change is acceptable since it will not adversely affect plant operations and will require two RHR pumps to be operable during Mode 6 when the water level is below 20 ft and above the flange. CTS do not address these conditions. Since Mode 6 includes the activities when the refueling cavity is being flooded or drained, a new NOTE 2 is included to allow the RHR pumps to be aligned to the RWST during these evolutions. NOTE 2 will also apply during control rod unlatching and latching operations and during upper internals removal and replacement. Therefore, the specific exception for these activities is not included in ITS. Since these changes place more restrictions on plant operations, they are more restrictive changes. These changes are consistent with the guidance of NUREG-1431.
	38	Not used.
	39	Not used.
	40	Not used.
	41	Not used.
	42	Not used.

NSHD Category	Change Number 3.9-	Discussion of Change
M	43	New SRs, 3.9.6.1 and 3.9.6.2 are included which require verification that one RHR loop is in operation and verify proper breaker alignment and power available to the other RHR pump. These new SRs are simple observations of plant conditions and will not adversely impact plant operations. Since these will be formal TS required surveillances, this change is considered more restrictive. This change is included to make the PI ITS complete and consistent with the guidance of NUREG-1431.
LR	44	CTS 3.8.A.1.h. The requirement for communication between the control room and containment is not included. No screening criteria apply for this requirement since communications is not part of the primary success path assumed in mitigation of a DBA or transient. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is acceptable since the TRM is under the controls of 10CFR50.59. This change is consistent with the guidance of NUREG-1431.
	45	Not used.
LR	46	3.8.A.1.i. To be consistent with the guidance of NUREG-1431, the CTS restriction on moving fuel prior to 100 hours after the reactor is subcritical is not included in the ITS. This requirement will be relocated to the TRM. This change is acceptable since plant refueling preparations take longer than 100 hours, thus it is not possible to move fuel prior to this time. Since this restriction will not be a TS requirement, this is a less restrictive change.

NSHD Category	Change Number 3.9-	Discussion of Change
A	47	3.8.A.2 and 3.8.A.3. Clarification is added to the Action Statements to incorporate the Applicability changes for each individual specification which were discussed previously. Since each change was previously discussed, these changes are considered administrative.
A	48	CTS 3.8.A.3. When specification requirements are not met, CTS require "no reduction in reactor coolant boron concentration shall be made." To consistent with NUREG-1431 as modified by TSTF-286, this CTS requirement is clarified by the addition of the clause, "less than required to meet LCO 3.9.1." Since the Specification requirements (shutdown boron concentrations) continue to be met, this change does not substantively change plant operations. Thus this is an administrative change.
L	49	Table 4.1-2B, Test 8 and Note 3. This change will require boron concentration verification every three days rather than the CTS requirement for daily verification. This change is acceptable due to experience with refueling boron concentration, the large volume of water, and the normal procedural controls of valve positions which make unplanned boron concentration changes unlikely.

PACKAGE 3.9

REFUELING OPERATIONS

PART E

MARKUP OF NUREG-1431
IMPROVED STANDARD TECHNICAL SPECIFICATIONS
AND BASES

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

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 limits specified in the COLR. CL3.9-51 CL3.9-60

APPLICABILITY: MODE 6.

TA3.9-52

NOTE
Only applicable to the refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limits.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
	AND		CL3.9-60
	A.3	Initiate action to restore boron concentration to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY CL3.9-60
SR 3.9.1.1	Verify boron concentration is within the limits specified in COLR.	72 hours

~~3.9 REFUELING OPERATIONS~~

TA3.9-53

~~3.9.2 Unborated Water Source Isolation Valves~~

~~LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.~~

~~APPLICABILITY: MODE 6.~~

~~ACTIONS~~

~~NOTE~~
~~Separate Condition entry is allowed for each unborated water source isolation valve.~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. NOTE Required Action A.3 must be completed whenever Condition A is entered.~~ ~~One or more valves not secured in closed position.~~	~~A.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ ~~A.2 Initiate actions to secure valve in closed position.~~ ~~AND~~ ~~A.3 Perform SR 3.9.1.1.~~	~~Immediately~~ ~~Immediately~~ ~~4 hours~~

~~SURVEILLANCE REQUIREMENTS~~

TA3.9-53

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.9.2.1~~	~~Verify each valve that isolates unborated water sources is secured in the closed position.~~	~~31 days~~

3.9 REFUELING OPERATIONS

3.9.~~2~~7 Refueling Cavity Water Level

PA3.9-54

LCO 3.9.~~2~~7 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.~~ TA3.9-66
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~~ TA3.9-66
	~~AND~~	
	~~A.2~~ Suspend movement of irradiated fuel assemblies within containment.	Immediately
	~~AND~~	
	~~A.3 Initiate action to restore refueling cavity water level to within limit.~~	~~Immediately~~ TA3.9-56

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.~~7~~2.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two ~~source range~~ neutron flux monitors and one neutron flux monitor audible countrate circuit shall be OPERABLE. CL3.9-57 CL3.9-58

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~[~~required~~] source range~~ neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately CL3.9-57
	AND	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1~~positive reactivity additions~~.	Immediately TA3.9-94

CONDITION	REQUIRED ACTION		COMPLETION TIME
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 CL3.9-57
	AND		
	B.2	Perform SR 3.9.1.1.	~~4 hours~~ TA3.9-59 ~~AND~~ Once per 12 hours ~~thereafter~~
C. Neutron flux monitor audible countrate circuit inoperable.	C.1	Suspend CORE ALTERATIONS.	Immediately CL3.9-58
	AND		
	C.2	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately PA3.9-70
	AND		
	C.3	Perform SR 3.9.1.1	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	X3.9-61 24~~[18]~~ months

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by ~~[~~four~~]~~ bolts;

b. One door in each air lock closed, or capable of being closed with at least two containment fan coil unit fans capable of operating in high speed; and CL3.9-62

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, or blind flange~~, or equivalent,~~ or CL3.9-65

2. capable of being closed by an OPERABLE Containment Ventilation ~~Purge and Exhaust~~ Isolation System. PA3.9-64

APPLICABILITY: ~~During CORE ALTERATIONS,~~ TA3.9-66
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 ~~Suspend CORE ALTERATIONS.~~	Immediately
	~~AND~~	TA3.9-66
	~~A.2~~ Suspend movement of irradiated fuel assemblies within containment.	~~Immediately~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	NOTE Not required to be met for containment inservice (low flow) purge valve(s) in penetrations closed to comply with LCO 3.9.4.c.1. Verify each required containment inservice (low flow) purge system ~~and exhaust~~ valve actuates to the isolation position on an actual or simulated actuation signal.	TA3.9-67 PA3.9-64 X3.9-61 24 ~~[18]~~ months

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

----------NOTE----------

The required RHR loop may not be in~~removed from~~ operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into~~reduction of~~ the Reactor Coolant System; coolant with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1. TA3.9-68 TA3.9-94

APPLICABILITY: MODE 6 with the water level ≥ 20~~23~~ ft above the top of reactor vessel flange. CL3.9-73

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RHR loop requirements not met.	A.1	Suspend operations that would cause introduction into the RCS, coolant with ~~involving a reduction in reactor coolant~~ boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately TA3.9-94 Immediately
	AND		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.3	Initiate action to satisfy RHR loop requirements.	
	AND		

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close equipment hatch and secure with four bolts~~all containment penetrations providing direct access from containment atmosphere to outside atmosphere~~.	4 hours TA3.9-69
	AND		
	A.5	Close one door in each air lock.	4 hours
	AND		
	A.6.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, or blind flange.	4 hours
	OR		
	A.6.2	Verify each inservice (low flow) purge penetration is capable of being closed by an OPERABLE Containment Ventilation Isolation System.	4 hours

ACTIONS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation~~ and circulating reactor coolant at a flow rate of ≥ [2800] gpm~~.	12 hours CL3.9-71

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

NOTES

1. The required operating RHR loop may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System (RCS), coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1. CL3.9-74

2. The required operating RHR loop may be removed from operation, and considered OPERABLE and in operation, to support filling and draining the reactor cavity when aligned to, or during transitioning to or from, the refueling water storage tank provided the required RHR loop is capable of being realigned to the RCS. X3.9-76

APPLICABILITY: MODE 6 with the water level < 20~~23~~ ft above the top of reactor vessel flange. CL3.9-73

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loop[s] to OPERABLE status.	Immediately X3.9-78
	OR	CL3.9-73
	A.2 Initiate action to establish ≥ 2~~0~~3 ft of water above the top of reactor vessel flange.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR loop in operation.	B.1	Suspend operations that would cause introduction into the RCS, coolant with ~~involving a reduction in reactor coolant~~ boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately TA3.9-94
	AND		
	B.2	Initiate action to restore one RHR loop to operation.	Immediately
	AND		
	B.3	Close equipment hatch and secure with four boltsall containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours TA3.9-69
	AND		(continued)

ACTIONS CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Close one door in each air lock.	4 hours
	AND	TA3.9-69
	B.5.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, or blind flange.	4 hours
	OR	
	B.5.2 Verify each penetration is capable of being closed by an OPERABLE Containment Ventilation Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation ~~and circulating reactor coolant at a flow rate of ≥ [2800] gpm~~.	12 hours CL3.9-71

SURVEILLANCE		FREQUENCY
SR 3.9.6.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

PA3.9-77

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS)~~, the refueling canal,~~ and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling. CL3.9-51

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration and associated shutdown margin limits ~~is~~are specified in the COLR. The required boron concentration will vary depending on time in core life. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures. CL3.9-60

AEC GDC Criterion 27 (Ref. 1) ~~GDC 26 of 10 CFR 50, Appendix A,~~ requires that two independent reactivity control systems of different design principles be provided ~~(Ref. 1)~~. AEC GDC Criterion 29 (Ref. 1) requires at least one ~~One~~ of these systems must be capable of holding the reactor core subcritical under any ~~cold~~ conditions. The Chemical and Volume Control System (CVCS), Safety Injection (SI) and Residual Heat Removal (RHR) are ~~is~~ the systems capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration. CL3.9-81

(continued)

BASES

BACKGROUND (continued)

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The ~~refueling canal and the~~ refueling cavity is~~are~~ then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps. CL3.9-51

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added ~~concentrated~~ boric acid with the water~~ in the refueling canal~~. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS~~, the refueling canal,~~ and the refueling cavity above the appropriate COLR limits. CL3.9-51 CL3.9-82

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 limits specified in the COLR are~~ is~~ based on the core reactivity at one or more points in the ~~beginning of each~~ fuel cycle ~~(the end of refueling)~~ and include~~s~~ an uncertainty allowance. CL3.9-82 CL3.9-60

(continued)

BASES

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, an adequate ~~at least a 5% Δk/k~~ margin of safety is established during refueling. PA3.9-84

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. CL3.9-51

The analyzed ~~limiting~~ boron dilution accident requiring the highest boron concentration ~~analyzed~~ occurs in MODE 6~~5~~ (Ref. 2). ~~A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F."~~ CL3.9-60 TA3.9-83

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii) ~~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 limits specified in the COLR ensure~~s~~ that a core k_{eff} of CL3.9-51

(continued)

BASES

LCO (continued)

≤ 0.95 or other lower value based on the dilution analysis is maintained during fuel handling operations. CL3.9-60

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} ≤ 0.95 or a lower apropriate level based on the dilution analysis. CL3.9-60 Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - ~~T_{avg} > 200°F,~~" ~~and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F,"~~ TA3.9-83 ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling cavity when connected to the RCS. When the refueling cavity is isolated from the RCS, no potential path for boron dilution exists. TA3.9-52

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 concentration of any coolant volume in the RCS~~,~~ or the refueling ~~canal, or the refueling~~ cavity, CL3.9-51 when connected, is less than that needed to maintain shutdown margin within its limit, all CL3.9-60

(continued)

BASES

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.

Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action. TP3.9-94

A.3

In addition to immediately suspending CORE ALTERATIONS and ~~or~~ positive reactivity additions, boration to restore the concentration must be initiated immediately. TP3.9-94

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.

ACTIONS

A.3 (continued)

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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of ~~the refueling canal, and~~ the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. CL3.9-51 TA3.9-52

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. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27 and 29, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~Appendix A, GDC 26~~. CL3.9-81

2. UFSAR, Chapter 14.4~~[15]~~. CL3.9-85

B 3.9 REFUELING OPERATIONS

B 3.9.2~~7~~ Refueling Cavity Water Level PA3.9-77

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 TA3.9-66 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 CL3.9-51 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 < 25% of 10 CFR 100 limits~~, as provided by the guidance of Reference 3~~. CL3.9-86

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS and~~ movement of irradiated fuel TA3.9-66 assemblies, the water level in the ~~refueling canal and the~~ refueling cavity is an initial condition CL3.9-51 design parameter in the analysis of a fuel handling accident in containment, as postulated by ~~Regulatory Guide 1.25 (Reference 1)~~. A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

(continued)

BASES

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 100 hours prior to fuel handling, 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 (Refs.3 and 4 ~~and 5~~). CL3.9-86

APPLICABLE SAFETY ANALYSES (continued)

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii) ~~the NRC Policy Statement~~.

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 3~~. CL3.9-86

APPLICABILITY

LCO 3.9.2~~7~~ 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. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.15, "Spent Fuel Storage Pool Water Level." TA3.9-66 CL3.9-87

(continued)

BASES

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. TA3.9-66

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

TA3.9-56

~~A.3~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.~~7~~2.1

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.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. ~~U~~FSAR, Section ~~14.~~5~~[15.4.5]~~. CL3.9-88

~~3. NUREG-0800, Section 15.7.4.~~ CL3.9-86

~~3~~4. 10 CFR 100.10.

~~4~~5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-~~7~~8828, Radiological Consequences of a Fuel Handling Accident, December 1971. CL3.9-89

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

PA3.9-77

BASES

BACKGROUND

~~The source range n~~Neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed ~~source range~~ neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors (N-31, N-32, N-51, and N-52) are located external to the reactor vessel and detect neutrons leaking from the core. CL3.9-57 CL3.9-90

The installed ~~source range~~ neutron flux monitors are: CL3.9-57

a. BF3 detectors operating in the proportional region of the gas filled detector characteristic curve; or CL3.9-90

b. Fission chambers.

The detectors monitor the neutron flux in counts per second. The instrument range used for monitoring changes in subcritical multiplication typically covers six decades of neutron flux ~~(1E+6 cps) with a [5]% instrument accuracy~~. The detectors ~~also~~ provide continuous visual indication in the control room. The installed BF3 neutron flux monitors provide ~~and~~ an audible indication~~alarm~~ to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1. CL3.9-91 CL3.9-58

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Two OPERABLE ~~source range~~ neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. ~~The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."~~ CL3.9-57 CL3.9-92

The ~~source range~~ neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii) ~~the NRC Policy Statement~~. CL3.9-57

LCO

This LCO requires that two ~~source range~~ neutron flux monitors, capable of monitoring subcritical neutron flux, be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. Neutron detectors N-31, N-32, N-51 and N-52 may be used to satisfy this LCO requirement. CL3.9-57

This LCO also requires that one audible countrate circuit, associated with either N-31 or N-32, be OPERABLE to ensure that audible indication is available to alert the operator in the event of a dilution accident. CL3.9-58

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~~se same~~ installed ~~source range~~ detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." CL3.9-57

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With only one required~~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 introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 ~~positive reactivity additions~~ must be suspended immediately. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not result in reducing SDM below the required value. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

CL3.9-57

TA3.9-94

TA3.9-94

PA3.9-97

B.1

With no ~~source range~~ required 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~~ required neutron flux monitor is restored to OPERABLE status.

CL3.9-57

(continued)

BASES

B.2

With no required~~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 that could lead to reducing SDM below the required value 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. CL3.9-57 TA3.9-94 CL3.9-57

The Completion Time of once per 12~~4~~ hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration~~.~~ and ~~The Frequency of once per 12 hours~~ ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low TA3.9-59

~~ACTIONS~~ ~~B.2 (continued)~~

probability of a change in core reactivity during this time period.

C.1, C.2 and C.3

With no audible neutron flux monitor countrate circuit OPERABLE, only visual indication is available and prompt and definite indication of a boron dilution event would be lost. Perfomance of Required Actions C.1 and C.2 shall not preclude completion of movement of a component to a safe position or that is a normal cooldown of the coolant volume for the purpose of system temperature control within established procedures. CL3.9-58 PA3.9-70

Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity is stabilized until the audible count rate capability is restored. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

BASES

The Completion Time of 12 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures 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. CL3.9-58

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every ~~18~~ 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. ~~The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data.~~ The ~~18~~24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed ~~at the 18 month Frequency~~. CL3.9-57 PA3.9-95 X3.9-61

BASES

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.~~ AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 13, 19, 27 and 31, issued for comment July 10, 1967, as referenced in USAR Section 1.2. CL3.9-81

2. U~~F~~SAR, Section 14.4~~[15.2.4]~~. CL3.9-93

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

PA3.9-77

BASES

BACKGROUND

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required. TA3.9-66

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least TA3.9-66

(continued)

BASES

four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

BACKGROUND (continued)

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, containment closure or closure capability is required; therefore, the door interlock mechanism may remain disabled and both doors may be open provided one door can be closed with at least two containment fan coil unit fans capable of operating in high speed~~, but one air lock door must always remain closed~~. TA3.9-66 CL3.9-62

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will ~~be restricted from escaping to the environment. The closure restrictions are sufficient to~~ restrict fission product radioactivity release from containment to be within regulatory limits~~due to a fuel handling accident during refueling~~. CL3.9-62 PA3.9-101

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 36~~42~~ inch purge penetration and a 36~~42~~ inch exhaust penetration. The second subsystem, a minipurge system referred to as containment

(continued)

BASES

inservice purge, includes ~~an~~ 48 inch purge penetration and an 8 inch exhaust penetration.

During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position, or the penetrations may be blank flanged. The two valves in each of the two containment inservice purge ~~minipurge~~ penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation System. ~~Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.~~ CL3.9-102 CL3.9-102

In MODE 6, sufficient air flow rates ~~large air exchangers~~ are necessary to conduct refueling operations. The ~~normal 42 inch~~ inservice purge system is used for this purpose, and ~~all~~ each of the four valves is ~~are~~ closed by the radiation monitors associated with the containment inservice purge system in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation." ~~ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."~~ The 36 inch subsystem is normally blank flanged, although the option for use is allowed during outages, except during movement of irradiated fuel or CORE ALTERATIONS. CL3.9-102 PA3.9-101

~~The minipurge system remains operational in MODE 6, and all four valves are also closed by the ESFAS. or The minipurge system is not used in MODE 6. All four 8 inch valves are secured in the closed position.~~ PA3.9-101

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one

(continued)

BASES

BACKGROUND (continued)

side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, or blind flange~~, or equivalent~~. ~~Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1)~~. CL3.9-65

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1~~2~~). Fuel handling accidents~~, analyzed in Reference 3,~~ include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.2~~7~~, "Refueling Cavity Water Level," in conjunction with~~and~~ the minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability~~CORE ALTERATIONS~~ ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. ~~Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.~~ The acceptance limit~~s~~ for offsite radiation exposure ~~will be~~is 25% of 10 CFR 100 values~~ or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits)~~.

CL3.9-86 PA3.9-54 TA3.9-66 CL3.9-62 CL3.9-86

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will restrict fission product relase from containment to be well within regulatory

CL3.9-62

(continued)

BASES

limits. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

A fuel handling accident does not cause containment pressurization; however, with an assumed single failure, the operating purge system supply fan is assumed to continue supplying air to containment. To maintain post-fuel handling accident releases well within the limits of 10CFR100, only the inservice purge system is allowed to be operating during fuel movement. Two fan coil unit fans are required to operate in the high speed mode following a fuel handling accident to assure that radioactive material in containment is well mixed and any releases will leave containment at a lower concentration over the duration of the accident. The provision that one air lock door is OPERABLE and under procedural control will assure that at least one door will be closed within 30 minutes as required, thus assuring radioactive releases are well within the limits of 10 CFR 100 (Ref. 1). CL3.9-62

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii) ~~the NRC Policy Statement~~.

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment.

The LCO requires ~~any~~ containment penetration~~s~~ ~~providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations.~~ to meet the following requirements: CL3.9-62

(continued)

BASES

a. The equipment hatch is closed and held in place by at least 4 bolts; CL3.9-62

b. One door in each air lock is closed, or capable of being closed with at least two containment fan coil unit fans capable of operating in high speed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either: PA3.9-64

1. closed by a manual or automatic isolation valve, or blind flange, or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System. PA3.9-64

The containment air lock doors may be open during movement of irradiated fuel in the containment provided that one door is capable of being closed within 30 minutes in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one door in each air lock will be closed following an evacuation of containment. CL3.9-62 CL3.9-104

For the OPERABLE containment inservice purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Ventilation Isolation System. ~~Containment Purge and Exhaust Isolation System.~~ The OPERABILITY requirements for this LCO require~~ensure~~ that the automatic containment inservice purge and exhaust valve closure times specified in the IST Program~~FSAR~~ can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

(continued)

BASES

APPLICABILITY

The containment penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment because this is when there is a potential for the limiting~~a~~ fuel handling accident. PA3.9-64

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment is ~~are~~ not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

~~A.1 and A.2~~ TA3.9-66

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment ~~Purge and Exhaust~~ Ventilation Isolation System not capable of automatic actuation when the inservice 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 ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a fuel assembly~~component~~ to a safe position. TA3.9-66 TA3.9-66

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open inservice purge and exhaust valves will demonstrate that the valves will function if required during a fuel handling accident~~are not blocked from closing~~. Also the Surveillance will

CL3.9-105

TA3.9-66

TA3.9-66

SURVEILLANCE REQUIREMENTS

~~SR 3.9.4.1 (continued)~~

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic Ccontainment Ventilation~~purge and exhaust~~ Iisolation signal.

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance is to be conducted before the start of refueling operation and then in accordance with the frequency specified~~will provide two or three surveillance verifications during the applicable period for this LCO~~. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment.

PA3.9-64

TA3.9-66

(continued)

BASES

SR 3.9.4.2

This Surveillance demonstrates that each containment inservice purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24~~18~~ month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.5~~6~~, the Containment Ventilation ~~Purge and Exhaust~~ Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24~~18~~ months a CHANNEL CALIBRATION is performed. ~~The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS.~~ SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances, when performed, ~~during MODE 6~~ will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

PA3.9-64

X3.9-61

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

TA3.9-67

(continued)

BASES (continued)

REFERENCES

1. ~~GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~ CL3.9-65

2. ~~UFSAR, Section 14.5[15.4.5].~~ CL3.9-88

3. ~~NUREG-0800, Section 15.7.4, Rev. 1, July 1981.~~ CL3.9-86

B 3.9 REFUELING OPERATIONS PA3.9-77

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS)~~, as required by GDC 34~~, to provide mixing of borated coolant and to prevent boron stratification~~ (Ref. 1)~~. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg~~(s)~~. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

CL3.9-108

CL3.9-109

PA3.9-110

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel~~. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant~~ which would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level

PA3.9-111

(continued)

BASES

CL3.9-73

≥ 20~~3~~ ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.~~

~~The RHR System, during refueling conditions, satisfies Criterion 4 of 10CFR 50.36(c)(2)(ii).~~

PA3.9-113

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 20~~3~~ ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability.

CL3.9-73

At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes a~~n~~ RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature.

(continued)

BASES

The flow path starts in one of the RCS hot legs and is returned to a~~the~~ RCS cold leg~~s~~.

PA3.9-110

The LCO is modified by a Note that allows the required operating RHR loop to not be ~~removed from~~ in operation~~service~~ for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute ~~cause a reduction of~~ the RCS boron concentration with coolant at boron concentrations less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction, with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained, is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

TA3.9-68

TA3.9-94

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 20~~3~~ ft above the top of the reactor vessel flange, to provide decay heat removal. ~~The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level."~~

CL3.9-73

~~APPLICABILITY (continued)~~

Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and

(continued)

BASES

Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 20~~3~~ ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level." CL3.9-73

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated.~~ TA3.9-94

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not result in reducing core reactivity below the required SDM or refueling boron concentration limit. TA3.9-94 PA3.9-97

(continued)

BASES

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 20~~3~~ ft above the reactor vessel flange provides an adequate available heat sink. CL3.9-73

Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

CL3.9-55

In accordance with LCO 3.9.2, "Refueling Cavity Water Level," movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling

ACTIONS ~~A.3 (continued)~~

water level ≥ 20~~3~~ ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately. CL3.9-73

A.4, A.5, A.6.1, and A.6.2

TA3.9-69

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts;

BASES

b. One door in each air lock must be closed; and

TA3.9-69

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, or blind flange, or verified to be capable of being closed by an OPERABLE Containment Ventilation Isolation System.

CL3.9-65

~~If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours.~~ With the RHR loop requirements not met, the potential exists for the coolant to boil, clad to fail, and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all~~Closing~~ containment penetrations are either closed or can be closed so that~~ are open to the outside atmosphere ensures~~ the dose limits are not exceeded.

TA3.9-69

PA3.9-127

The Completion Time of 4 hours ~~is reasonable, based on the low probability of the coolant boiling in that time.~~ allows adequate time to fulfill the Required Actions and not exceed dose limits.

PA3.9-116

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation in order ~~and circulating reactor coolant. The flow rate is determined by the flow rate necessary~~ to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available

CL3.9-71

BASES

to the operator in the control room for monitoring the RHR System.

CL3.9-109

REFERENCES ~~1. Section FSAR, [5.5.7]~~None.

B 3.9 REFUELING OPERATIONS

PA3.9-77

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS)~~, as required by GDC 34~~, to provide mixing of borated coolant, and to prevent boron stratification ~~(Ref. 1)~~. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg~~(s)~~. Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

CL3.9-108

CL3.9-109

PA3.9-110

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel~~. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will~~ , which could eventually challenge the integrity of the fuel cladding, ~~which is~~ a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

PA3.9-111

(continued)

BASES

~~Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.~~ PA3.9-113

The RHR System, during refueling conditions, satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODE 6, with the water level < 2~~0~~3 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. CL3.9-73

LCO (continued)

Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold leg~~s~~. PA3.9-110

Either RHR pump may be aligned to the Refueling Water Storage Tank (RWST) to support filling or draining the refueling cavity or for performance of required testing. PA3.9-122

(continued)

BASES

The LCO contains two Notes which provide clarification of the LCO. CL3.9-74

Note 1 allows the required operating RHR loop to not be in service for up to 1 hour per 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration with coolant at concentrations less than required to meet the minimum boron concentration of LCO 3.9.1. The RHR loop only needs to be running if needed.

Note 2 allows a required RHR loop to be removed from operation, but considered OPERABLE and in operation during filling or draining of the refueling cavity, when aligned to or during transitioning to or from the RWST, provided the required RHR loop is capable of being realigned to the RCS. This variance allows the unit to perform necessary evolutions and recognizes that the RHR loop is still functional and that the system can be realigned easily. This note allows the full flow test of RHR to the upper head injection path. X3.9-76

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 20~~3~~ ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 20~~3~~ ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level." CL3.9-73

(continued)

BASES

ACTIONS

A.1 and A.2

If less than the required number of RHR loop~~(s)~~ are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 20~~3~~ ft of water level is established above the reactor vessel flange. When the water level is ≥ 20~~3~~ ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

X3.9-78

CL3.9-73

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.~~ Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not

TA3.9-94

PA3.9-97

(continued)

BASES

result in reducing core reactivity below the required SDM or refueling boron concentration limit.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B5.1 and B.5.2

TA3.9-69

If no RHR loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with four bolts;

b. One door in each air lock must be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, or blind flange, or verified to be capable of being closed by an OPERABLE Containment Ventilation Isolation System.

CL3.9-65

~~If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours.~~ With the RHR loop requirements not met, the potential exists for the coolant to boil, clad to fail, and release radioactive gas to the containment

TA3.9-69

PA3.9-127

(continued)

BASES

atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded. ~~Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.~~ TA3.9-69

The Completion Time of 4 hours ~~is reasonable, based on the low probability of the coolant boiling in that time.~~ allows adequate time to fulfill the Required Actions and not exceed dose limits. PA3.9-116

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant~~. The flow rate is determined by the flow rate necessary~~ to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. CL3.9-71

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control,

SURVEILLANCE REQUIREMENTS

~~SR 3.9.6.1 (continued)~~

and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.6.2 X3.9-124

Verification that the required pump is OPERABLE ensures that an additional ~~RCS or~~ RHR pump can be placed in operation, if needed, to maintain decay heat removal and ~~reactor coolant~~ circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered CL3.9-125

BASES

reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES ~~1. FSAR, Section [5.5.7]~~None. CL3.9-109

PACKAGE 3.9

REFUELING OPERATIONS

PART F

JUSTIFICATION FOR DIFFERENCES
(JFD)

from

NUREG-1431
IMPROVED STANDARD TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART F

PACKAGE 3.9

REFUELING OPERATIONS

JUSTIFICATION FOR DIFFERENCES FROM IMPROVED STANDARD TECHNICAL SPECIFICATIONS (NUREG-1431) AND BASES

See Part E for specific proposed wording and location of referenced deviations.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	51	The term "refueling canal" has been deleted from this specification and the Bases since PI does not have a plant design feature by this name. PI has a refueling cavity, fuel transfer tube and fuel transfer canal. The fuel transfer canal is outside of containment and is not included in this specification.
TA	52	Incorporates TSTF-272, Revision 1. This change clarifies that the refueling cavity boron concentration is only required to meet this Specification when it is relied upon to maintain core reactivity. In particular, if the refueling cavity is not connected to the RCS or is not filled with water, it would not be expected to meet this Specification. The implementation of this TSTF has been modified to read correctly by deleting "the refueling canal" since PI does not have a design feature by this name.

Difference Category	Difference Number 3.9-	Justification for Differences
TA	53	Incorporated TSTF-23, Rev 3. PI has analyzed the boron dilution accident and concluded that there is adequate time for operator action to mitigate the event. Therefore this Specification is not applicable to PI and is not included.
PA	54	NUREG-1431 Specification 3.9.7 has been renumbered as PI ITS Specification 3.9.2 to maintain the numbering of subsequent Specifications without leaving a Specification number vacancy due to inapplicability of the Unborated Water Source Isolation Valves specification at PI.
CL	55	Clarification is provided that the water level of 23 ft is required for handling fuel, even though this Required Action may apply when the water level is less than 23 ft but above 20 feet. If the water level is below 23 feet, fuel is not being handled and Required Action A.2 would not apply.
TA	56	Incorporates TSTF-20.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	57	The term "source range" has not been included because PI uses other neutron flux detectors (gammametrics) in this function in addition to the source range neutron flux monitors. The use of the more general term "neutron flux monitors" is consistent with CTS which allows use of monitors other than the source range neutron flux monitors. Since there are more than two monitors which could be used, "required" is retained in the Action Statements.
CL	58	An LCO statement, action statement and supporting Bases requiring an operable audible neutron flux countrate circuit are included to retain the CTS requirements. The audible countrate circuit is necessary because PI depends on operator action to mitigate the consequences of a boron dilution event. The installed source range neutron detectors are the instruments which provide the audible coutrate. Also, approved TSTF-23, Rev 3 requires audible countrate indication and an associated action statement. The logical connector "AND" format is not used since this is not consistent with the description of Logical Connectors provided in ISTS Section 1.2.
TA	59	Incorporates approved traveler TSTF-96, Revision 1.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	60	The COLR will specify multiple boron concentration limits for varying refueling conditions when the ITS is implemented. Thus, "limits" is specified as plural.
X	61	This ITS conversion LAR proposes to allow the PI refueling outages on a 24 month cycle. Accordingly the surveillance interval on this new SR is revised to 24 months.
CL	62	CTS LA 119/112 allows CORE ALTERATIONS with the containment airlock doors open. Changes to NUREG-1431 Specifications and Bases preserve this capability and include the CTS requirement for two containment fan coil unit fans to be capable of operating in high speed. This change is also consistent with the intent of approved TSTF-68 Revision 2. This LAR demonstrated that the fission product release from the containment following a fuel handling accident will be within regulatory limits under the assumed containment penetration and airlock status.
	63	Not used.

Difference Category	Difference Number 3.9-	Justification for Differences
PA	64	The PI name for the instrumentation system which automatically isolates containment ventilation during fuel handling is the "Containment Ventilation Isolation System" and the Specification for this system is 3.3.5, "Containment Ventilation Isolation Instrumentation". The system which is isolated is the "containment inservice (low flow) purge system", thus this name is used in SR 3.9.4.2 and throughout the Bases. The parenthetical modifier "(low flow)" may be included to assure that the operators do not confuse this system with the containment purge system.
	65	Not Used.

Difference Category	Difference Number 3.9-	Justification for Differences
TA	66	This change incorporates approved TSTF-51, Revision 2. The accidents postulated to occur during core alteration, in addition to fuel handling accidents, are: inadvertant criticality (due to a control rod removal error or continuous rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with a fuel assembly in an improper location. These events have been analyzed at PI and do not result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, this Specification is revised to only apply during fuel handling. This TSTF also includes an option of limiting the Specification to only apply to "recently discharged" irradiated fuel. PI has not performed the calculations to support this limitation and thus this option is not included with the incorporation of this TSTF.
TA	67	This change incorporates TSTF-284, Revision 3.
TA	68	This change incorporates TSTF-153.
TA	69	Incorporates TSTF-197, Revision 2. This change provides more definitive guidance to the operators for the actions which must be taken. The changes also include plant specific terminology for further clarification. See PA3.9-116 for further discussion of exceptions to TSTF-197.

Difference Category	Difference Number 3.9-	Justification for Differences
PA	70	New Required Actions, C.2 and C.3, have been included to be consistent with the intent of TSTF-268, Revision 2 which was not considered when TSTF-23 was written.
CL	71	SR 3.9.5.1 and SR 3.9.6.1 and their Bases were revised to remove the flow rate for the RHR loop in operation. The PI safety analysis for boron dilution in MODE 6 assumes uniform mixing of the borated coolant as a result of a RHR pump being in operation and does not specify a flow rate. Therefore, there is no basis for inclusion of a flow rate in the SR. The phrase "and circulating reactor coolant" was not included since this is an implied function for an RHR loop in operation and is consistent the safety analysis. This change is also consistent with the guidance of the letter to Mr. James Davis, NEI from William D. Beckner, NRC, dated April 29, 1999,
	72	Not used.
CL	73	The water level in 3.9.5 and 3.9.6 Applicability Statements below which two RHR pumps are required to be operable has been changed to 20 ft to retain the CTS requirement. The level of 23 ft as specified in NUREG-1431 is not practical at PI since it does not allow sufficient operating flexibility. The 23 ft level in NUREG-1431, per the Bases, was selected as a matter of convenience. (See change A3.9-32 for further discussion). This change has also been made in the Bases.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	74	CTS allows both RHR pumps to not be operating regardless of the refueling cavity water level to allow plant refueling evolutions. A LCO Note is included in ITS LCO 3.9.6 to preserve this CTS allowed operating condition. Associated Bases for this Note have also been provided.
	75	Not used.

Difference Category	Difference Number 3.9-	Justification for Differences
X	76	ITS LCO 3.9.6 requires two RHR loops OPERABLE and one loop in operation. This change would modify the LCO and Bases by a Note that allows one required RHR loop to be removed from operation and considered OPERABLE to allow alignment to, or during transitioning to or from, the RWST to support filling the reactor cavity if the required RHR loop is capable of being aligned to the RCS. This change is necessary as the required flow path for the RHR loop is from the RCS hot leg, through the RHR pump and heat exchanger, to the RCS cold leg. However, in order to fill the refueling cavity through the RCS hot legs; and, in order to drain the refueling cavity, the suction of the RHR pump(s) is aligned to the RCS hot legs and the water is pumped into the RWST. CTS does not address this situation because CTS does not have specification requirements when the water level is above the reactor vessel flange and CORE ALTERATIONS or fuel movements are ot being performed. The allowance provided by the Note is acceptable because the conditions for which the Note applies are strictly controlled evolutions performed in accordance with plant procedures, the planning of these evolutions takes into consideration unit conditions and the impact on shutdown risk, and the Note is only applicable when the required RHR loop is capable of being realigned to the RCS.

Difference Category	Difference Number 3.9-	Justification for Differences
PA	77	Included throughout the Bases are reference corrections, renumbering and relettering of paragraphs and minor wording changes which have been made to accommodate changes to the Specifications and Prairie Island (PI) unique needs. These changes are not identified by change numbers.
X	78	The action statement, as written, requires entry when only one RHR loop is inoperable, in which case, only a single RHR loop would require restoration. Thus, the plural "s" on loops is put in parentheses. This change has also been made in the Bases as appropriate.
	79	Not used.
	80	Not used.
CL	81	As discussed in USAR, Section 1, Prairie Island license basis uses the Criteria proposed in AEC General Design Criteria for Nuclear Power Plant Construction Permits, issued for comment July 10, 1967.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	82	The COLR specifies limits based on an analysis of the dilution accident in MODE 6 which may differ from other plants. Accordingly, changes have been made to the Bases which account for PI analyses. For example, the term "appropriate" is used in the Background and "and is conservative" has been removed from the Applicable Safety Analyses.
TA	83	These changes implement TSTF-9 and TSTF-136 which corrects Shutdown Margin section to 3.1.1 and title to reflect that actually used in Section 3.1.
PA	84	The use of 5% $\Delta K/K$ is not consistent with $K_{eff} \leq 0.95$, is not consistent with our training of operators and therefore may confuse the operators. Since ≤ 0.95 adequately defines the situation, 5% $\Delta K/K$ is not included.
CL	85	USAR Section 14 is the correct reference for the dilution analysis.
CL	86	The dose limit of 25% of 10 CFR 100 is based on the current licensing basis of License Amendments 119/112. NSP originally committed to Safety Guide 25 (current Regulatory Guide 1.25). Since NUREG-0800 does not apply to PI it has not been included as a reference.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	87	The title of LCO 3.7.15 is "Spent Fuel Storage Water Level", consistent with plant terminology.
CL	88	Correct FSAR reference to applicable USAR section dealing with fuel handling accident.
CL	89	Correct WCAP reference to WCAP-7828.
CL	90	The PI neutron flux monitors are not limited to the BF3 design. The Gammametric detectors actually span the source, intermediate, and power ranges, thus the range and instrument accuracy statements are broadened to be applicable to the PI case. The Gammametric detectors are of fission chamber design.
CL	91	Clarification is provided to identify that the neutron flux monitor function is for monitoring changes in subcritical multiplication.
CL	92	A safety analysis for the boron dilution accident has been performed for PI, thus reference to the original NUREG-1431 section is incorrect and is eliminated.
CL	93	The correct PI reference is USAR Section 14.

Difference Category	Difference Number 3.9-	Justification for Differences
TA	94	Incorporates traveler TSTF-286, Revision 2. Changes have been made to the Bases discussion for clarification, accuracy and to make it read correctly. These changes are required to provide operator guidance on acceptable activities when Required Actions are entered.
PA	95	The details of CHANNEL CALIBRATION have not been included since this description is not accurate for PI and this level of detail is not necessary in the Technical Specification Bases.
CL	96	References 1 and 2 are changed to be consistent with licensing basis, specifically the AEC GDC and USAR, section 14.4.
PA	97	An additional sentence has been included with the clarification provided by TSTF-286. This additional sentences states, "Introduction of temperature changes, including temperature increases when operating with a positive moderator coefficient, must also be evaluated to not result in reducing SDM below the required value." This statement is borrowed from another insert provided in TSTF-286 and is appropriate for this Required Action at PI. This statement provides further guidance to the operator and assures that the plant will be operated in a conservative manner.
	98	Not used.

Difference Category	Difference Number 3.9-	Justification for Differences
	99	Not used.
	100	Not used.
PA	101	The PI containment purge and inservice purge systems have different dimensions than the ITS reference. The minipurge system is called the "inservice purge" system at Prairie Island. Since the containment purge system is seldom used, much of its discussion has been revised to apply to the inservice purge system and some of the minipurge system discussion has not been included.
PA	102	The PI containment purge system is normally blank flanged. The PI inservice purge system valves are isolated by 1R11, 1R12, or 1R22 [2R11, 2R12, or 2R22]. These radiation monitors are part of the Containment Ventilation Isolation instrumentation and are not considered part of the ESFAS. Reference to MODE 5 is irrelevant for the MODE 6 specification, thus it is not included.
	103	Not used.
CL	104	CTS requirements that an open airlock can be closed within 30 minutes following a fuel handling accident have been relocated to the Bases.

Difference Category	Difference Number 3.9-	Justification for Differences
CL	105	CTS allows the containment airlock doors to be open during handling of irradiated fuel and assumes that some radioactive releases may occur following a fuel handling accident. Thus "significant" is added to make this statement true. This change also incorporates TSTF-51.
	106	Not used.
	107	Not used.
CL	108	The PI licensing basis is AEC draft GDC. Those criteria do not have a criterion which specifically addresses decay heat removal as does 10 CFR 50, Appendix A. Thus reference to GDC 34 is not included.
CL	109	The PI USAR does not address boron stratification during refueling operations and thus this reference is not included.
PA	110	Prairie Island RHR System returns only to loop B, thus "leg" should only be singular.

Difference Category	Difference Number 3.9-	Justification for Differences
PA	111	The discussion of boiling would be true for a small volume, but is likely not an accurate description of events that would occur in the refueling cavity and reactor. Those portions of the discussion which are inaccurate for PI have not been included.
	112	Not used.
PA	113	The standard wording is deleted because we believe the RHR system during refueling condition meets criterion 4 of 10CFR 50.36, i.e. "A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."
	114	Not used.
	115	Not used.
PA	116	The Completion Time discussion is amplified to clarify the reasoning since it may not be true for PI as stated in NUREG-1431 as modified by approved TSTF-197, Revision 2.
	117	Not used.
	118	Not used.

Difference Category	Difference Number 3.9-	Justification for Differences
	119	Not used.
	120	Not used.
	121	Not used.
PA	122	Additional flow paths are provided to clarify the discussion of the subsequent NOTE.
	123	Not used.
X	124	This is an administrative change which corrects apparent format error whereby the SR 3.9.6.2 was not included in the original NUREG-1431.
CL	125	The verification required by this SR only relates to the RHR system, thus references to the RCS and reactor coolant are not included.
	126	Not used.
CL	127	A clarification is added "clad to fail" since radioactive gases of concern would not be released unless this occurs.

PACKAGE 3.9

REFUELING OPERATIONS

PART G

NO SIGNIFICANT HAZARDS DETERMINATION (NSHD)

and

ENVIRONMENTAL ASSESSMENT

for

CHANGES TO PRAIRIE ISLAND CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Part G

PACKAGE 3.9

REFUELING OPERATIONS

NO SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10CFR Part 50, Section 50.91 using the standards provided in Section 50.92.

For ease of review, the changes are evaluated in groupings according to the type of change involved. A single generic evaluation may suffice for some of the changes while others may require specific evaluation in which case the appropriate reference change numbers are provided.

A - Administrative (GENERIC NSHD)
(A3.9-02, A3.9-03, A3.9-05, A3.9-06, A3.9-07, A3.9-10, A3.9-12, A3.9-14, A3.9-26, A3.9-28, A3.9-32, A3.9-47, A3.9-48)

Most administrative changes have not been marked-up in the Current Technical Specifications, and may not be specifically referenced to a discussion of change. This No Significant Hazards Determination (NSHD) may be referenced in a discussion of change by the prefix "A" if the change is not obviously an administrative change and requires an explanation.

These proposed changes are editorial in nature. They involve reformatting, renaming, renumbering, or rewording of existing Technical Specifications to provide consistency with NUREG-1431 or conformance with the Writer's Guide, or change of current plant terminology to conform to NUREG-1431. Some administrative changes involve relocation of requirements within the Technical Specifications without affecting their technical content. Clarifications within the new Prairie Island Improved Technical

A – Administrative (continued)

Specifications which do not impose new requirements on plant operation are also considered administrative.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed conversion of Prairie Island Current Technical Specifications to conform to NUREG-1431 involves reformatting, rewording, changes in terminology and relocating requirements. These changes are simply editorial, or do not involve technical changes and thus they do not impact any initiators of previously analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed administrative changes do not involve physical modification of the plant, no new or different type of equipment will be installed or removed associated with these administrative changes, nor will there be changes in parameters governing normal plant operation. The proposed administrative changes do not impose new or different requirements on plant operation. Therefore, these administrative changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   These proposed administrative changes do not impact any safety analysis assumptions. Therefore, these changes do not involve a reduction in the plant margin of safety.

M - More restrictive (GENERIC NSHD)
(M3.9-08, M3.9-16, M3.9-21, M3.9-22, M3.9-23, M3.9-24, M3.9-33, M3.9-36, M3.9-37, M3.9-43)

This proposed Technical Specifications revision involves modifying the Current Technical Specifications to impose more stringent requirements upon plant operations to achieve consistency with the guidance of NUREG-1431, correct discrepancies or remove ambiguities from the specifications. These more restrictive Technical Specifications have been evaluated against the plant design, safety analyses, and other Technical Specifications requirements to ensure the plant will continue to operate safely with these more stringent specifications.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes provide more stringent requirements for operation of the plant. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event.

   These more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed changes do not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed, nor do they change the methods governing normal plant operation.

   These more stringent requirements do impose different operating restrictions. However, these operating restrictions are consistent with the boundaries established by the assumptions made in the plant safety analyses and licensing bases. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

M - More restrictive

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The imposition of more stringent requirements on plant operation either has no impact on the plant margin of safety or increases the margin of safety. Each change in this category is by definition providing additional restrictions to enhance plant safety by:

a) increasing the analytical or safety limit;
b) increasing the scope of the specifications to include additional plant equipment;
c) adding requirements to current specifications;
d) increasing the applicability of the specification;
e) providing additional actions;
f) decreasing restoration times;
g) imposing new surveillances; or
h) decreasing surveillance intervals.

These changes maintain requirements within the plant safety analyses and licensing bases. Therefore, these changes do not involve a significant reduction in a margin of safety.

R - Relocation (GENERIC NSHD)
(None in this Package)

This License Amendment Request (LAR) proposes to relocate requirements contained in the Current Technical Specifications out of the Technical Specifications into licensee controlled programs. These requirements are relocated because they 1) do not meet the Technical Specifications selection criteria defined in 10 CFR 50.36; or 2) are mandated by current Nuclear Regulatory Commission (NRC) regulations and are therefore unnecessary in the Technical Specifications.

In the NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (dated 7/16/93), the NRC stated:

> . . . since 1969, there has been a trend towards including in Technical Specifications not only those requirements derived from the analyses and evaluations included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power reactors. . . This has contributed to the volume of Technical Specifications and to the several-fold increase, since 1969, in the number of license amendment applications to effect changes to the Technical Specifications. It has diverted both staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety.

Thus, relocation of unnecessary requirements from the Current Technical Specifications should result in an overall improvement in plant safety through more focused attention to the requirements that are most important to plant safety.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   These proposed changes relocate requirements for structures, systems, components or variables which did not meet the criteria for inclusion in the improved Technical Specifications or duplicate regulatory requirements. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events.

   These relocated operability requirements will continue to be maintained pursuant to 10 CFR 50.59, other regulatory requirements (as applicable for the document

R - Relocation (continued)

to which the requirement is relocated), or the Administrative Controls section of these proposed improved Technical Specifications.

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes do not impose any different requirements and adequate control of existing requirements will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

These proposed changes will not reduce the margin of safety because they do not impact any safety analysis assumptions. In addition, the relocated requirements for the affected structure, system, component or variables are the same as the current Technical Specifications. Since future changes to these requirements will be evaluated per the requirements of 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Controls section of the Improved Technical Specifications, proper controls are in place to maintain the plant margin of safety. Therefore, these changes do not involve a significant reduction in the margin of safety.

LR - Less restrictive, Relocated details (GENERIC NSHD)
(LR3.9-01, LR3.9-11, LR3.9-13, LR3.9-17, LR3.9-18, LR3.9-27, LR3.9-44, LR3.9-46)

Some information in the Prairie Island Current Technical Specifications that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and relocated to the proposed Bases, Updated Safety Analysis Report or licensee controlled procedures. The relocation of this descriptive information to the Bases of the Improved Technical Specifications, Updated Safety Analysis Report or licensee controlled procedures is acceptable because these documents will be controlled by the Improved Technical Specifications required programs, procedures or 10CFR50.59. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes relocate detailed, descriptive requirements from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures. These documents containing the relocated requirements will be maintained under the provisions of 10CFR50.59, a program or procedure based on 10CFR50.59 evaluation of changes, or NRC approved methodologies. Since these documents to which the Technical Specifications requirements have been relocated are evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC approval. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not necessitate physical alteration of the plant; that is, no new or different type of equipment will be installed, or change parameters governing normal plant operation. The proposed changes will not impose any different requirements and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

LR - Less restrictive, Relocated details (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addiction, the requirements to be transposed from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures are the same as the existing Technical Specifications. Since future changes to these requirements will be evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no reduction in a margin of safety will be allowed without prior NRC approval. Therefore, these changes do not involve a significant reduction in a margin of safety.

L - Less restrictive, Specific

Each CTS change which is designated as Less (L prefix) restrictive on plant operations is provided with a specific NSHD.

Specific NSHD for Change L3.9-04

CTS specify the status of containment penetrations during CORE ALTERATIONS to limit radioactive releases following an accident. The accidents of concern during CORE ALTERATIONS are fuel handing accidents, inadvertent criticality and misloading a fuel assembly. Of these accidents, only fuel handling accidents have potential for significant fuel cladding integrity damage and subsequent radioactive release. Therefore it is acceptable to specify containment penetration status only during fuel handling activities.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change involves narrowing the activities during which containment penetration status is controlled by TS. Since containment penetrations are not accident initiators, these changes do not affect the probability of an accident. Fuel cladding damage is not postulated to occur from CORE ALTERATION accidents, other than fuel handling accidents; therefore removing TS on containment penetrations during these accidents does not involve a significant increase in the consequences of an accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

Specific NSHD for Change L3.9-04 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change does not involve a significant reduction in margin of safety because the activities no longer covered by this specification are not postulated to release radioactive materials.

Therefore it is concluded this proposed change does not involve a significant hazards consideration.

Specific NSHD for Change L3.9-31

CTS require the refueling cavity water level be verified prior to moving fuel or control rods. The proposed ITS will specify that the refueling cavity water level be verified at least every 24 hours. This change is acceptable because of the large volume of water and normal procedural controls of valve positions which make significant unplanned level changes unlikely. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change removes the requirement to verify refueling cavity level prior to moving fuel or control rods, but rather relies upon the periodic verification every 24 hours. Since the water level is not an accident initiator this change does not increase the probability of an accident. The general rules of use and applicability for ITS (SR 3.0.4) require that all surveillance requirements be successfully met and current prior to entering the Mode of Applicability. Performance of the surveillance requirements confirming the refueling cavity water level prior to entering the Mode of Applicability ensures that the assumptions of the safety analysis are met and does not affect the performance of any credited equipment. Therefore, no analyses assumptions are violated and there is no significant increase in the consequences of a previously analyzed event.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Relaxing the requirement to confirm the refueling cavity water level just prior to entering the Mode of Applicability does not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. Thus, this change does not create the possibility of a new or different kind of accident.

Specific NSHD for Change L3.9-31 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigation actions are initiated. Relaxing the requirement to verify the refueling cavity water level prior to entering the Mode of Applicability does not significantly impact these factors. There are no design changes or equipment performance parameter changes associated with this change. No setpoints are affected, and no change is being proposed in the plant operation limits as a result of this change. Thus, relaxing the requirement to confirm the refueling cavity water level parameter prior to entering the Mode of Applicability does not involve a significant reduction in the plant margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.9-34

CTS allow the one required operating RHR pump to be shut down for up to one hour "to facilitate movement of fuel or core components" during CORE ALTERATIONS. PI ITS requires one RHR loop to be operating during MODE 6 with shutdown of the loop allowed for one hour per eight hour period providing operations which could reduce the boron concentration are not performed. The specific activities for which the pump may be shutdown are not specified. Thus this change may allow the RHR loop to be shut down for other reasons, for example, latching and unlatching of control rods. This change is acceptable because the time during which the loop is out of service is limited and adequate controls are provided to assure that the plant is maintained in a safe condition. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change allows the operating RHR loop to be shut down without specifying the purposes for which it may be shut down. This change does not change the length of time which the RHR loop may be shut down and therefore it does not involve a significant increase in the probability of an accident previously evaluated. This change does not alter the concept of allowing the RHR system to be totally inoperable for some period of time during refueling operations; thus it does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

Specific NSHD for Change L3.9-34 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   This change removes the specific purposes for which the operating RHR loop many be shut down; however, it does not change the amount of time which the loop may be shut down. This change does include further limitations that the loop may be shut down only one hour per eight hour period and operations which could reduce the boron concentration are not permitted. Thus, this change allows the operating RHR loop to be shut down for the same amount of time for any purpose with further restrictions. Overall this change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.9-49

The proposed change would require verification of refueling cavity boron concentration every three days rather than daily as required by CTS. This change is acceptable due to experience with refueling boron concentration, the large volume of water, and the normal procedural controls of valve positions which make unplanned boron concentration changes unlikely. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change increases the surveillance interval to 72 hours to verify refueling cavity boron concentration. Unplanned boron concentration changes are unlikely due to the procedural control on valve positions for water supplies to the cavity. Also, the refueling cavity contains a large quantity of water; thus any changes in boron concentration will occur slowly. Therefore, this change does not involve a significant increase in the probability of a previously analyzed accident. The consequences of a boron dilution accident are not affected by the change in surveillance interval; therefore this change does not involve a significant increase the consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Specific NSHD for Change L3.9-49 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The valves which provide water to the refueling cavity are under procedural controls which make cavity boron dilution unlikely. Furthermore, due to the large volume of water, boron dilution would occur slowly. Therefore the proposed change does not result in a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

ENVIRONMENTAL ASSESSMENT

The Nuclear Management Company has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.

PACKAGE 3.9

REFUELING OPERATIONS

CROSS - REFERENCE

CURRENT TECHNICAL SPECIFICATIONS

TO

IMPROVED TECHNICAL SPECIFICATIONS

List of Section Cross - References

3.8
Table 1-1
Table 4.1-2A
Table 4.1-2B

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.8				
3.8.A.1		LCO	3.9.4	
3.8.A.1.a.1		LCO	3.9.4	
3.8.A.1.a.2.a		LCO	3.9.4	
3.8.A.1.a.2.b.i			Relocated - TRM	
3.8.A.1.a.2.b.ii		LCO	3.9.4	
3.8.A.1.a.2.b.iii			Relocated - Bases	
3.8.A.1.a.2.b.iv		LCO	3.9.4	
New		SR	3.9.4.1	
New		SR	3.9.4.2	
3.8.A.1.b			Relocated - TRM	
3.8.A.1.c		LCO	3.9.3	
New		LCO	3.9.3	
3.8.A.1.c		(Partial)	Relocated - Bases	
New		SR	3.9.3.1	
New		SR	3.9.3.2	
3.8.A.1.d		LCO	3.9	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.8.A.1.e		LCO	3.9.2	
3.8.A.1.e		SR	3.9.2.1	
3.8.A.1.e		(Partial)	Relocated - Bases	
3.8.A.1.f		LCO	3.9.5	
3.8.A.1.f		LCO	3.9.6	
New		SR	3.9.5.1	
3.8.A.1.g		LCO	3.9.6	
New		LCO	3.9.6	
New		SR	3.9.6.1	
New		SR	3.9.6.2	
3.8.A.1.h			Relocated - TRM	
3.8.A.1.i			Relocated - TRM	
3.8.A.1.j		LCO	3.3.5	
3.8.A.2		LCO	3.9.2	
3.8.A.2		LCO	3.9.3	
3.8.A.2		LCO	3.9.4	
3.8.A.2		LCO	3.9.5	
3.8.A.2		LCO	3.9.6	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
3.8.A.3		LCO	3.9.5	
3.8.A.3		LCO	3.9.6	
3.8.B.1		LCO	3.7.16	
3.8.B.1.a			Relocated - TRM	
3.8.B.1.b			Relocated - TRM	
3.8.B.1.c		LCO	3.7.16	
3.8.B.2		LCO	3.7.16	
3.8.C			Relocated - TRM	
3.8.D.1		LCO	3.7.13	
3.8.D.2		LCO	3.7.13	
New		LCO	3.7.13	
3.8.D.3		LCO	3.7.13	
3.8.D.4		LCO	3.7.13	
New		LCO	3.7.15	
New		SR	3.7.15.1	
3.8.E.1		LCO	3.7.17	
3.8.E.2		LCO	3.7.16	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section Table				
Table 1-1		TABLE	Table 1.1-1	
Table 1-1	Note *	LCO	3.9.1	
New		LCO	3.9.1	
Table 1-1	Note *	(Partial)	Relocated - COLR	
Table 1-1	Note **		Deleted	
Table 3.5-1	9	TABLE	3.3.5-1	Note c
Table 3.5-1	1	TABLE	3.3.2-1	1c
Table 3.5-1	2a	TABLE	3.3.2-1	2c
Table 3.5-1	2b	TABLE	3.3.2-1	4b
Table 3.5-1	3	TABLE	3.3.2-1	1d
Table 3.5-1	4	TABLE	3.3.2-1	1e
Table 3.5-1	4	TABLE	3.3.2-1	Note b
Table 3.5-1	5	TABLE	3.3.2-1	4c
Table 3.5-1	6	TABLE	3.3.2-1	4d
Table 3.5-1	7	SR	3.6.8.1	
Table 3.5-1	8		Relocated - TRM	
Table 3.5-1	9	TABLE	3.3.5-1	3

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-1	10	SR	3.3.4.2	
Table 3.5-2A	1	TABLE	3.3.1-1	1
Table 3.5-2A	2a	TABLE	3.3.1-1	2a
Table 3.5-2A	2b	TABLE	3.3.1-1	2b
Table 3.5-2A	3	TABLE	3.3.1-1	3a
Table 3.5-2A	4	TABLE	3.3.1-1	3b
Table 3.5-2A	5	TABLE	3.3.1-1	4
Table 3.5-2A	6	TABLE	3.3.1-1	5
Table 3.5-2A	7	TABLE	3.3.1-1	6
Table 3.5-2A	8	TABLE	3.3.1-1	7
Table 3.5-2A	9	TABLE	3.3.1-1	8a
Table 3.5-2A	10	TABLE	3.3.1-1	8b
Table 3.5-2A	11	TABLE	3.3.1-1	9
Table 3.5-2A	12	TABLE	3.3.1-1	10
Table 3.5-2A	13	TABLE	3.3.1-1	14
Table 3.5-2A	14	TABLE	3.3.1-1	13
Table 3.5-2A	15	TABLE	3.3.1-1	12
Table 3.5-2A	16a	TABLE	3.3.1-1	11a
Table 3.5-2A	16b	TABLE	3.3.1-1	11b

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2A	17	TABLE	3.3.1-1	15
Table 3.5-2A	18	TABLE	3.3.1-1	19
Table 3.5-2A	19	TABLE	3.3.1-1	17
Table 3.5-2A	20	TABLE	3.3.1-1	17
Table 3.5-2A	New Func	TABLE	3.3.1-1	16
Table 3.5-2A	New Func	TABLE	3.3.1-1	18
Table 3.5-2A	Act 1	LCO	3.3.1 B	
Table 3.5-2A	Action 1	LCO	3.3.1 M	
Table 3.5-2A	Action 2	LCO	3.3.1 D	
Table 3.5-2A	Action 2	LCO	3.3.1 E	
Table 3.5-2A	Act 2	SR	3.2.4.2	
Table 3.5-2A	Act 2c	SR	3.2.4.2	
Table 3.5-2A	Act 3	LCO	3.3.1 F	
Table 3.5-2A	New Action	LCO	3.3.1 G	
Table 3.5-2A	Action 4	LCO	3.3.1 H	
Table 3.5-2A	New Action	LCO	3.3.1 I	
Table 3.5-2A	Action 5	LCO	3.3.1 J	
Table 3.5-2A	Action 6	LCO	3.3.1 E	
Table 3.5-2A	Action 6	LCO	3.3.1 K	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2A	Action 6	LCO	3.3.1 N	
Table 3.5-2A	Action 7	LCO	3.3.1 O	
Table 3.5-2A	Act 8	LCO	3.3.1 C	
Table 3.5-2A	Action 9a	LCO	3.3.1 S	
Table 3.5-2A	Action 9a	LCO	3.3.1.P	
Table 3.5-2A	Action 9b	LCO	3.3.1 P	
Table 3.5-2A	Action 10	LCO	3.3.1 C	
Table 3.5-2A	Act 10	LCO	3.3.1 P	
Table 3.5-2A	Action11	LCO	3.3.1 L	
Table 3.5-2A	New Action	LCO	3.3.1 Q	
Table 3.5-2A	New Action	LCO	3.3.1 R	
Table 3.5-2A	New Action	LCO	3.3.1 S	
Table 3.5-2A	Note a	TABLE	3.3.1-1	Note a
Table 3.5-2A	Note b	TABLE	3.3.1-1	Note b
Table 3.5-2A	Note c	TABLE	3.3.1-1	Note d
Table 3.5-2A	Note d	TABLE	3.3.1-1	Note i
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note e
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note f
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note g

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note h
Table 3.5-2A	New Note	TABLE	3.3.1-1	Note j
Table 3.5-2B	1a	TABLE	3.3.2-1	1a
Table 3.5-2B	1b	TABLE	3.3.2-1	1c
Table 3.5-2B	1c	TABLE	3.3.2-1	1e
Table 3.5-2B	1d	TABLE	3.3.2-1	1d
Table 3.5-2B	1e	TABLE	3.3.2-1	1b
Table 3.5-2B	2a	TABLE	3.3.2-1	2a
Table 3.5-2B	2b	TABLE	3.3.2-1	2c
Table 3.5-2B	2c	TABLE	3.3.2-1	2b
Table 3.5-2B	3a	TABLE	3.3.2-1	3c
Table 3.5-2B	3b	TABLE	3.3.2-1	3a
Table 3.5-2B	3c	TABLE	3.3.2-1	3b
Table 3.5-2B	4a	TABLE	3.3.5-1	5
Table 3.5-2B	4b	TABLE	3.3.5-1	1
Table 3.5-2B	4c	TABLE	3.3.5-1	6
Table 3.5-2B	4d	TABLE	3.3.5-1	4
Table 3.5-2B	4e	TABLE	3.3.5-1	3
Table 3.5-2B	4f	TABLE	3.3.5-1	2

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2B	5a	LCO	3.7.2	
Table 3.5-2B	5b	TABLE	3.3.2-1	4b
Table 3.5-2B	5c	TABLE	3.3.2-1	4d
Table 3.5-2B	5d	TABLE	Not used	
Table 3.5-2B	5e	TABLE	3.3.2-1	4a
Table 3.5-2B	6a	TABLE	3.3.2-1	5b
Table 3.5-2B	6b	TABLE	3.3.2-1	5c
Table 3.5-2B	6c		Relocated - TRM	
Table 3.5-2B	6d	TABLE	3.3.2-1	5a
Table 3.5-2B	7a		Relocated - TRM	
Table 3.5-2B	7b	TABLE	3.3.2-1	6b
Table 3.5-2B	7c	TABLE	3.3.2-1	6d
Table 3.5-2B	7c	TABLE	3.3.2-1	Note f
Table 3.5-2B	7d	TABLE	3.3.2-1	6e
Table 3.5-2B	7d*	TABLE	3.3.2-1	Note g
Table 3.5-2B	7e	TABLE	3.3.2-1	6c
Table 3.5-2B	7f	TABLE	3.3.2-1	6a
Table 3.5-2B	8a	LCO	3.3.4.a	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2B	8b	LCO	3.3.4.b	
Table 3.5-2B	9		Deleted - LAR	
Table 3.5-2B	Act 20	LCO	3.3.2 C	
Table 3.5-2B	Act 21	LCO	3.3.2 D	
Table 3.5-2B	Act 21	LCO	3.3.2 E	
Table 3.5-2B	Act 22	LCO	3.3.5 A	
Table 3.5-2B	Act 23	LCO	3.3.2 B	
Table 3.5-2B	Act 24	LCO	3.3.2 D	
Table 3.5-2B	Act 24	LCO	3.3.2 G	
Table 3.5-2B	Act 25	LCO	3.3.2 F	
Table 3.5-2B	Act 26	LCO	3.3.2 I	
Table 3.5-2B	Act 27	LCO	3.7.2	
Table 3.5-2B	Act 28	LCO	3.3.2 F	
Table 3.5-2B	Act 29	LCO	3.3.2 D	
Table 3.5-2B	Act 29	LCO	3.3.2 H	
Table 3.5-2B	Act 30	LCO	3.3.2 I	
Table 3.5-2B	Act 31	LCO	3.3.4 A	
Table 3.5-2B	Act 32		Deleted	
Table 3.5-2B	Act 33	LCO	3.3.4 B	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.5-2B	Act 34		Deleted - LAR	
Table 3.5-2B	New Action	LCO	3.3.4 C	
Table 3.5-2B	New Action	LCO	3.3.4 D	
Table 3.5-2B	Act 35		Deleted - LAR	
Table 3.5-2B	Act 36		Deleted - LAR	
Table 3.5-2B	Note a	TABLE	3.3.2-1	Note a
Table 3.5-2B	Note b	TABLE	3.3.5-1	Note a, b
Table 3.5-2B	Note c	TABLE	3.3.2-1	Note c
Table 3.5-2B	Note c	LCO	3.7.2	
Table 3.5-2B	Note d	TABLE	3.3.2-1	Note c,d
Table 3.5-2B	New Note	TABLE	3.3.2-1	Note e
Table 3.15-1	1	TABLE	3.3.3-1	1
Table 3.15-1	2	TABLE	3.3.3-1	2
Table 3.15-1	3	TABLE	3.3.3-1	3
Table 3.15-1	4	TABLE	3.3.3-1	4
Table 3.15-1	5	TABLE	3.3.3-1	5
Table 3.15-1	6	TABLE	3.3.3-1	6
Table 3.15-1	7	TABLE	3.3.3-1	7
Table 3.15-1	8	TABLE	3.3.3-1	8

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.15-1	9	TABLE	3.3.3-1	9
Table 3.15-1	10	TABLE	3.3.3-1	10
Table 3.15-1	11	TABLE	3.3.3-1	11
Table 3.15-1	12	TABLE	3.3.3-1	12
Table 3.15-1	13	TABLE	3.3.3-1	13
Table 3.15-1	14	TABLE	3.3.3-1	14
Table 3.15-1	15	TABLE	3.3.3-1	15
Table 3.15-1	16	TABLE	3.3.3-1	16
Table 3.15-1	Action a	LCO	3.3.3	
Table 3.15-1	Action a1	LCO	3.3.3 A	
Table 3.15-1	Action a1	LCO	3.3.3 C	
Table 3.15-1	Action a2	LCO	3.3.3 D	
Table 3.15-1	Action a2	LCO	3.3.3 I	
Table 3.15-1	Action a3	LCO	3.3.3 D	
Table 3.15-1	Action a3	LCO	3.3.3 J	
Table 3.15-1	Action a4	LCO	3.3.3 E	
Table 3.15-1	Action a4	LCO	3.3.3 I	
Table 3.15-1	Action a5	LCO	3.3.3 B	
Table 3.15-1	Action a5	LCO	3.3.3 C	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 3.15-1	Action a5	LCO	3.3.3	
Table 3.15-1	Action a6	LCO	3.3.3 F	
Table 3.15-1	Action a6	LCO	3.3.3 G	
Table 3.15-1	Action a6	LCO	3.3.3 I	
Table 3.15-1	New Cond	LCO	3.3.3 H	
Table 3.15-1	Action b	TABLE	3.3.3-1	Note a
Table 3.15-1	Action c	TABLE	3.3.3-1	Note b
Table 3.15-1	New Note	TABLE	3.3.3-1	Note c
Table 4.1-1A	1	TABLE	3.3.1-1	1
Table 4.1-1A	2a	TABLE	3.3.1-1	2a
Table 4.1-1A	2a	TABLE	3.3.1-1	6
Table 4.1-1A	2a	TABLE	3.3.1-1	7
Table 4.1-1A	2b	TABLE	3.3.1-1	2b
Table 4.1-1A	3	TABLE	3.3.1-1	3a
Table 4.1-1A	4	TABLE	3.3.1-1	3b
Table 4.1-1A	5	TABLE	3.3.1-1	4
Table 4.1-1A	6	TABLE	3.3.1-1	5
Table 4.1-1A	7	TABLE	3.3.1-1	6
Table 4.1-1A	8	TABLE	3.3.1-1	7

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1A	9	TABLE	3.3.1-1	8a
Table 4.1-1A	10	TABLE	3.3.1-1	8b
Table 4.1-1A	11	TABLE	3.3.1-1	9
Table 4.1-1A	12	TABLE	3.3.1-1	10
Table 4.1-1A	13	TABLE	3.3.1-1	14
Table 4.1-1A	14	TABLE	3.3.1-1	13
Table 4.1-1A	15	TABLE	3.3.1-1	12
Table 4.1-1A	16a	TABLE	3.3.1-1	11a
Table 4.1-1A	16b	TABLE	3.3.1-1	11b
Table 4.1-1A	17	TABLE	3.3.1-1	15
Table 4.1-1A	18	TABLE	3.3.1-1	19
Table 4.1-1A	19	TABLE	3.3.1-1	17
Table 4.1-1A	20	TABLE	3.3.1-1	17
Table 4.1-1A	New Func	TABLE	3.3.1-1	16
Table 4.1-1A	New Func	TABLE	3.3.1-1	18
Table 4.1-1A	Note 1	TABLE	3.3.1-1	Note a
Table 4.1-1A	Note 2	TABLE	3.3.1-1	Note d
Table 4.1-1A	Note 3	TABLE	3.3.1-1	Note b
Table 4.1-1A	Note 4	SR	3.3.1.8	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1A	Note 4a	SR	3.3.1.15	
Table 4.1-1A	Note 5	SR	3.3.1.2	
Table 4.1-1A	Note 6	SR	3.3.1.3	
Table 4.1-1A	Note 7	SR	3.3.1.3	
Table 4.1-1A	Note 7	SR	3.3.1.11	
Table 4.1-1A	Note 8	SR	3.3.1.6	
Table 4.1-1A	Note 9	SR	3.3.1.4	
Table 4.1-1A	Note 9	SR	3.3.1.5	
Table 4.1-1A	Note 10	SR	3.3.1.8	
Table 4.1-1A	Note 10	(Partial)	Relocated - Bases	
Table 4.1-1A	Note 11	SR	3.3.1.9	
Table 4.1-1A	Note 11	SR	3.3.1.15	
Table 4.1-1A	Note 12	TABLE	3.3.1-1	18
Table 4.1-1A	Note 13		Relocated - Bases	
Table 4.1-1A	Note 14		Relocated - Bases	
Table 4.1-1A	Note 15	TABLE	3.3.1-1	17
Table 4.1-1A	Note 16	TABLE	3.3.1-1	Note i
Table 4.1-1A	New Note	SR	3.3.1.4	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1A	Note 17	SR	3.3.1.8	
Table 4.1-1A	Note 18		Relocated - TRM	
Table 4.1-1A	New Note	SR	3.3.1.16	
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note c
Table 4.1-1A	New Note	SR	3.3.1.16	
Table 4.1-1A	New Note	SR	3.3.1.10	
Table 4.1-1A	New Note	SR	3.3.1.11	
Table 4.1-1A	New Note	SR	3.3.1.12	
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note e
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note f
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note g
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note h
Table 4.1-1A	New Note	TABLE	3.3.1-1	Note j
Table 4.1-1B	1a	TABLE	3.3.2-1	1a
Table 4.1-1B	1b	TABLE	3.3.2-1	1c
Table 4.1-1B	1c	TABLE	3.3.2-1	1e
Table 4.1-1B	1d	TABLE	3.3.2-1	1d
Table 4.1-1B	1e	TABLE	3.3.2-1	1b

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1B	2a	TABLE	3.3.2-1	2a
Table 4.1-1B	2b	TABLE	3.3.2-1	2c
Table 4.1-1B	2c	TABLE	3.3.2-1	2b
Table 4.1-1B	3a	TABLE	3.3.2-1	3c
Table 4.1-1B	3b	TABLE	3.3.2-1	3a
Table 4.1-1B	3c	TABLE	3.3.2-1	3b
Table 4.1-1B	4a	TABLE	3.3.5-1	5
Table 4.1-1B	4b	TABLE	3.3.5-1	1
Table 4.1-1B	4b	SR	3.3.5.4	
Table 4.1-1B	4c	TABLE	3.3.5-1	6
Table 4.1-1B	4d	TABLE	3.3.5-1	4
Table 4.1-1B	4e	TABLE	3.3.5-1	3
Table 4.1-1B	4e	SR	3.3.5.1	
Table 4.1-1B	4e	SR	3.3.5.3	
Table 4.1-1B	4e	SR	3.3.5.5	
Table 4.1-1B	4f	TABLE	3.3.5-1	2
Table 4.1-1B	4f	SR	3.3.5.2	
Table 4.1-1B	5a	SR	3.7.2.1	
Table 4.1-1B	5a	(partial)	Relocated - IST	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1B	5b	TABLE	3.3.2-1	4b
Table 4.1-1B	5c	TABLE	3.3.2-1	4d
Table 4.1-1B	5d	TABLE	3.3.2-1	4c
Table 4.1-1B	5e	TABLE	3.3.2-1	4a
Table 4.1-1B	6a	TABLE	3.3.2-1	5b
Table 4.1-1B	6b	TABLE	3.3.2-1	5c
Table 4.1-1B	6c		Relocated - TRM	
Table 4.1-1B	6d	TABLE	3.3.2-1	5a
Table 4.1-1B	7a		Relocated - TRM	
Table 4.1-1B	7b	TABLE	3.3.2-1	6b
Table 4.1-1B	7c	TABLE	3.3.2-1	6d
Table 4.1-1B	7c	TABLE	3.3.2-1	Note f
Table 4.1-1B	7d	TABLE	3.3.2-1	6e
Table 4.1-1B	7e	TABLE	3.3.2-1	6c
Table 4.1-1B	7f	TABLE	3.3.2-1	6a
Table 4.1-1B	8	SR	3.3.4.2	
Table 4.1-1B	8	SR	3.3.4.1	
Table 4.1-1B	Note 20	SR	3.3.2.5	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1B	Note 21	TABLE	3.3.2-1	Note a
Table 4.1-1B	Note 22	SR	3.3.2.2	
Table 4.1-1B	Note 23	TABLE	3.3.2-1	Note c
Table 4.1-1B	Note 23	LCO	3.7.2	
Table 4.1-1B	Note 24	TABLE	3.3.5-1	Note d
Table 4.1-1B	Note 25		Deleted	
Table 4.1-1B	Note 26	LCO	3.3.5-1	
Table 4.1-1B	New Note	TABLE	3.3.2-1	Note e
Table 4.1-1B	7d	TABLE	3.3.2-1	Note g
Table 4.1-1C	1		Relocated - TRM	
Table 4.1-1C	2	SR	3.1.4.1	
Table 4.1-1C	2	SR	3.1.7.1	
Table 4.1-1C	2	(Partial)	Relocated - TRM	
Table 4.1-1C	2	(Partial)	Deleted	
Table 4.1-1C	3		Relocated - TRM	
Table 4.1-1C	4		Relocated - TRM	
Table 4.1-1C	5		Deleted - Boric Acid LAR	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	6		Relocated - TRM	
Table 4.1-1C	7		Deleted - Boric Acid LAR	
Table 4.1-1C	8	SR	3.3.3.1	
Table 4.1-1C	8	SR	3.3.3.2	
Table 4.1-1C	9		Deleted - Boric Acid LAR	
Table 4.1-1C	10	SR	3.6.8.1	
Table 4.1-1C	10	SR	3.6.8.2	
Table 4.1-1C	11	SR	3.3.4.1	
Table 4.1-1C	12		Deleted - Boric Acid LAR	
Table 4.1-1C	13		Relocated - TRM	
Table 4.1-1C	14		CTS Deleted	
Table 4.1-1C	15		Relocated - TRM	
Table 4.1-1C	16		Relocated - TRM	
Table 4.1-1C	17		Relocated - TRM	
Table 4.1-1C	18	SR	3.3.1.12	
Table 4.1-1C	19		Relocated - TRM	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	20		Relocated - TRM	
Table 4.1-1C	21	SR	3.3.3.1	
Table 4.1-1C	21	SR	3.3.3.2	
Table 4.1-1C	21	SR	3.3.3.3	
Table 4.1-1C	22		CTS Deleted	
Table 4.1-1C	23		CTS Deleted	
Table 4.1-1C	24		Relocated - TRM	
Table 4.1-1C	24	SR	3.3.6.5	
Table 4.1-1C	24	SR	3.3.6.2	
Table 4.1-1C	25	SR	3.4.12.4	
Table 4.1-1C	25	SR	3.4.12.5	
Table 4.1-1C	25	SR	3.4.13.5	
Table 4.1-1C	25	SR	3.4.13.6	
Table 4.1-1C	26		Relocated - TRM	
Table 4.1-1C	27		Relocated - TRM	
Table 4.1-1C	28		Relocated - TRM	
Table 4.1-1C	29	SR	3.3.3.1	

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CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-1C	29	SR	3.3.3.2	
Table 4.1-1C	29	(Partial)	Relocated - TRM	
Table 4.1-1C	30		Relocated - TRM	
Table 4.1-1C	31		Relocated - TRM	
Table 4.1-1C	Note 30	SR	3.1.7.1	
Table 4.1-1C	Note 31		Deleted	
Table 4.1-1C	Note 32		Relocated - TRM	
Table 4.1-1C	Note 33		Deleted - Boric Acid LAR	
Table 4.1-1C	Note 34		Deleted	
Table 4.1-1C	Note 35		Deleted	
Table 4.1-1C	Note 36		Deleted	
Table 4.1-1C	Note 37		Deleted	
Table 4.1-1C	Note 38	SR	3.4.12.4	
Table 4.1-1C	Note 38	SR	3.4.13.5	
Table 4.1-1C	Note 39	SR	3.6.8.2	
Table 4.1-1C	Note 39	SR	3.6.8.1	
Table 4.1-1C	New Note	SR	3.3.3.3	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-2A	1	SR	3.1.4.3	
Table 4.1-2A	1	(Partial)	Relocated - TRM	
Table 4.1-2A	2	SR	3.1.4.2	
Table 4.1-2A	3	SR	3.4.10.1	
Table 4.1-2A	4	SR	3.7.1.1	
Table 4.1-2A	5	SR	3.9.2.1	
Table 4.1-2A	6	SR	3.4.11.1	
Table 4.1-2A	7	SR	3.4.11.2	
Table 4.1-2A	8		CTS Deleted	
Table 4.1-2A	9	SR	3.4.14.1	
Table 4.1-2A	10		CTS Deleted	
Table 4.1-2A	11		Relocated - TRM	
Table 4.1-2B	1	SR	3.4.17.1	
Table 4.1-2B	2	SR	3.4.17.2	
Table 4.1-2B	3	SR	3.4.17.3	
Table 4.1-2B	4a	LCO	3.4.17	
Table 4.1-2B	4b	SR	3.4.17.2	
Table 4.1-2B	5		Relocated - TRM	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-2B	6		Relocated - TRM	
Table 4.1-2B	7		Deleted in CTS	
Table 4.1-2B	8		Relocated - TRM	
Table 4.1-2B	8	SR	3.9.1.1	
Table 4.1-2B	9	SR	3.5.4.2	
Table 4.1-2B	10		Deleted by Boric Acid LAR	
Table 4.1-2B	11	SR	3.6.6.3	
Table 4.1-2B	12	SR	3.5.1.4	
Table 4.1-2B	13	SR	3.7.16.1	
Table 4.1-2B	14		Relocated - TRM	
Table 4.1-2B	15	SR	3.7.14.1	
Table 4.1-2B	16		Relocated - TRM	
Table 4.1-2B	Note 1	SR	3.4.17.3	
Table 4.1-2B	Note 2		Relocated - TRM	
Table 4.1-2B	Note 3	SR	3.9.1.1	
Table 4.1-2B	Note 4		Relocated - TRM	
Table 4.1-2B	Note 5		Deleted	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
Table 4.1-2B	Note 6		Relocated - TRM	
Table 4.2-1	1	G	5.5.6	
Table 4.12-1		G	5.5.8	
Table 4.12-2		G	5.5.8	
Table 4.13-1			Relocated - TRM	

PACAKGE 3.9

REFUELING OPERATIONS

CROSS - REFERENCE

IMPROVED TECHNICAL SPECIFICATIONS

TO

CURRENT TECHNICAL SPECIFICATIONS

Section Cross - Reference

Section 3.9

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
ITS Section 3.9				
3.9.1		LCO	Table 1-1	Note *
3.9.1		LCO	New	
3.9.1.1		SR	Table 4.1-2B	8
3.9.1.1		SR	Table 4.1-2B	Note 3
3.9.2		LCO	3.8.A.1.e	
3.9.2		LCO	3.8.A.2	
3.9.2.1		SR	3.8.A.1.e	
3.9.2.1		SR	Table 4.1-2A	5
3.9.3		LCO	3.8.A.1.c	
3.9.3		LCO	3.8.A.2	
3.9.3		LCO	New	
3.9.3.1		SR	New	
3.9.3.2		SR	New	
3.9.4		LCO	3.8.A.1	
3.9.4		LCO	3.8.A.2	
3.9.4.1		SR	New	
3.9.4.2		SR	New	

Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
3.9.5		LCO	3.8.A.1.f	
3.9.5		LCO	3.8.A.2	
3.9.5		LCO	3.8.A.3	
3.9.5		LCO	New	
3.9.5.1		SR	New	
3.9.6		LCO	3.8.A.1.f	
3.9.6		LCO	3.8.A.1.g	
3.9.6		LCO	New	
3.9.6		LCO	3.8.A.2	
3.9.6		LCO	3.8.A.3	
3.9.6.1		SR	New	
3.9.6.2		SR	New	