NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

MAY 07 2007

L-PI-07-018
10 CFR 50.71(e)
TS 5.5.12

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Updated Safety Analysis Report (USAR) Revision 29 and Bases Revisions

Pursuant to the applicable regulations and Technical Specifications (TS), the Nuclear Management Company, LLC (NMC) by this letter submits USAR page revisions and TS Bases page revisions for the Prairie Island Nuclear Generating Plant (PINGP).

Enclosure 1, Bases Page Changes, contains three copies of TS Bases, Revisions 181, 182, 183, and 184, page changes and instructions for entering the pages. These revisions are submitted pursuant to Technical Specification (TS) 5.5.12.d for TS Bases changes which have been implemented since the previous USAR submittal.

Enclosure 2, Information Regarding Changes to the USAR, identifies those changes made based on approved license amendments, changes made under the provisions of 10 CFR 50.59, 10 CFR 50.46, and editorial changes including deletion of particular information and the basis for that deletion.

Enclosure 3, Updated Safety Analysis Report, is a CD-ROM containing USAR Revision 29 in its entirety. This revision was made pursuant to 10 CFR 50.71(e), using the guidance of NEI 98-03, Rev. 1, "Guidelines for Updating Final Safety Analysis Reports." NMC requests that USAR Revision 28 be destroyed or marked superseded.

Summary of Commitments

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

A053

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Letter Number: L-PI-07-018 **Document Date:** 5/7/07

Date: **Engineer:** Kivi

Document Type: **Attached:**

Subject: USAR Rev 29

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Contact Marlys Davis at ext. 4154 if you did not receive what is indicated or to request a change to this distribution list.

I certify that the information presented herein accurately presents changes made to the Prairie Island USAR since the last updating submittal up through November 15, 2006.

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC (w/o Enclosures 1 and 3)

ENCLOSURE 1

BASES PAGE CHANGES

Updating Instructions (2 pages)

Revisions 181 through 184 Package (37 pages x 3 copies)

Updating Instructions

Remove and discard individual Bases pages and replace with the new pages provided. Special instructions, where applicable, are included with the replacement pages.

When page removal/replacement is complete, review the Bases Current Pages list to ensure your copy of the Bases is current and complete. Contact the Prairie Island Nuclear Generating Plant at 651-388-1121 if you require additional assistance.

BASES

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B RoR-12	9/1/05	Record of Revision	B RoR-12	5/24/06
			B RoR-13	9/14/06
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G	10/27/04		G	9/14/06
H	1/27/05		H	9/14/06
I	1/27/05		I	4/20/06

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RECORD OF REVISIONS
BASES CHANGES AND LICENSE AMENDMENTS

PI Revision (Rev) No.	Date of Issue	License Amendment No. DPR-42	DPR-60	Remarks
171	6/18/04	-	-	Correct Bases B 3.0.6 and add clarification that each steam generator is separate in B 3.3.3.
172	7/2/04	162	153	Incorporate Bases changes associated with transition to Westinghouse safety analysis and includes use of BEACON for core monitoring.
173	6/8/04	163	154	Remove H_2 recombiner and H_2 monitor discussions from Bases.
174	8/27/04	-	-	Revise 3.5.1 discussion of boron build up analyses.
175	9/10/04	166	156	Approved Alternate Source Term (AST) methodology for Fuel Handling Accident and implemented AST by revising TS and Bases B 3.3.5, B 3.9.2 and B 3.9.4.
176	10/27/04	167	157	Revised Bases 3.0, 3.1.3, 3.4.11, 3.4.12, 3.4.13, 3.4.16, 3.4.17, 3.5.3, 3.7.3, 3.7.4, 3.7.5 and 3.8.1 to incorporate LCO 3.0.4 flexibility changes.
177	10/27/04	-	-	Miscellaneous minor corrections to Bases 3.3.3 and 3.4.12.
178	1/27/05	-	-	Revise Bases 3.7.10 to require Air Handler for OPERABILITY and clarify fan test requirements, clarify SR 3.7.5.4 and restore B 3.3.3 discussion of SG Water Level channels
179	7/21/05	-	-	Revise Bases 3.3.7 Applicability to make it consistent with plant design and TS 3.3.7.
180	9/1/05	-	-	Revise Bases 3.8.4 due to revised minimum design battery voltage limit.
181	4/20/06	-	-	Miscellaneous clarifications in Bases 3.1.3, 3.6.5, 3.7.5 and 3.7.8.
182	2/5/06	172	162	Revised Bases due to revised fuel storage curves in TS 3.7.17 and TS 4.3 based on new criticality methodology and analyses.
183	5/24/06	-	-	Revised description of pressurizer heater supplies in SR 3.4.9.3.

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RECORD OF REVISIONS
BASES CHANGES AND LICENSE AMENDMENTS

PI Revision (Rev) No.	Date of Issue	License Amendment No DPR-42	DPR-60	Remarks
184	6/29/06	173	163	Modified TS Bases 3.6.5 to allow operation with any two containment fan coil units operable during the Completion Time. Revised TS Bases 3.7.8 to incorporate TS 3.6.5 changes.

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B 3.8.1-4	158/149
B 3.8.1-5	158/149
B 3.8.1-6	176/176
B 3.8.1-7	176/176
B 3.8.1-8	158/149
B 3.8.1-9	158/149
B 3.8.1-10	158/149
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B 3.8.1-13	158/149
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B 3.8.1-16	158/149
B 3.8.1-17	158/149
B 3.8.1-18	158/149
B 3.8.1-19	170
B 3.8.1-20	170
B 3.8.1-21	158/149
B 3.8.1-22	158/149
B 3.8.1-23	158/149

BASES

BACKGROUND (continued)	distributed poisons to yield an ITC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure that the ITC does not exceed the limits. The limitations on ITC in Limiting Condition for Operation (LCO) 3.1.3 ensure that the core is inherently stable during power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the reactor coolant system will compensate for any unintended reactivity increases. The limitations on ITC contained in the Core Operating Limits Report (COLR) are provided to ensure that the value of MTC remains within the limiting conditions assumed in the USAR accident and transient analyses. The operational upper limit of ITC (as specified in Condition A) is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO.
APPLICABLE SAFETY ANALYSES	The acceptance criteria for the specified ITC are: a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and b. The ITC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events. The USAR (Ref. 2) contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions for the cycle exposure

BASES

APPLICABLE SAFETY ANALYSES (continued)

being evaluated to ensure that the accident results are bounding.

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive (i.e., upper limit). Such accidents include the rod withdrawal transient from either zero or RTP, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include the main steam line break.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, ITC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the ITC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values will remain within the bounds of the original accident analyses during operation.

Assumptions made in safety analyses require that the ITC be less positive than a given upper bound and more positive than a given lower bound. The ITC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit usually occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the ITC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

BASES

LCO (continued)

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance check at BOC on ITC provides confirmation that the ITC is behaving as anticipated and will be within limits at 70% RTP, full power, and EOC so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

If the LCO limits are not met, the assumptions of the safety analysis may not be met. The core could violate criteria that prohibit a return to criticality, or the DNBR criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

APPLICABILITY

Technical Specifications place both LCO and SR values on ITC, based on the safety analysis assumptions described above.

In MODE 1, the limits on ITC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup accidents (such as the uncontrolled rod cluster control assembly withdrawal) will not violate the assumptions of the accident analysis. The lower ITC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents at EOC will not violate the assumptions of the accident analysis since ITC becomes more negative as the cycle burnup increases and the RCS boron concentration is reduced. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

BASES (continued)

ACTIONS

A.1

ITC must be kept within the upper limit specified in LCO 3.1.3 to ensure that assumptions made in the safety analysis remain valid. The upper limit of Condition A is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO.

If the upper ITC limit is violated at BOC, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits in the future. The ITC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the ITC measurement and computing the required bank withdrawal limits.

The control rods are maintained within the administrative withdrawal limits until a subsequent calculation verifies that ITC has been restored within its limit. As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the ITC to become more negative. Using physics calculations, the time in cycle life at which the calculated ITC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3 (continued)

This SR is not applicable for the Group A heaters since this group is permanently powered by a Class 1E power supply.

This Surveillance demonstrates that the Group B heaters can be manually transferred from the non-safeguards to the safeguards power supply and energized. The Frequency of 24 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.

REFERENCES

1. USAR, Section 14.
2. USAR, Section 4.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time incorporates delays to account for load restoration and motor windup (Ref. 3).

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a LOCA or SLB, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 4). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and thereby maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

Each Containment Spray System includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon a containment spray actuation signal. Manual valves in this system that could, if improperly positioned, reduce the spray flow below that assumed for accident analysis, are blocked and tagged in the proper position and maintained under administrative control. Containment Spray System motor operated valves, MV-32096 and MV-32097 (Unit 1), and MV-32108 and MV-32109 (Unit 2) are closed with the motor control center supply breakers in the off position.

Each Containment Cooling System typically includes cooling coils, dampers, fans, and controls to ensure an OPERABLE flow path. With one CL strainer isolated, the containment cooling train on the associated CL header is OPERABLE at CL supply temperatures

BASES

LCO (continued)	up to and including 70°F. When the CL supply temperature is above 70°F with one CL strainer isolated, the containment cooling train on the associated CL header is not OPERABLE. If Technical Specification (TS) 3.6.5 Condition D has been entered, then the above correlation between CL strainer status, CL supply temperature and containment cooling train OPERBILITY is not applicable. In this case the remaining two containment cooling fan coil units provide adequate heat removal within the TS 3.6.5 Condition D allowed Completion Time.
APPLICABILITY	In MODES 1, 2, 3, and 4, a LOCA or SLB could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.
ACTIONS	<u>A.1</u> With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the other Containment Spray train, reasonable time for repairs, and low probability of a LOCA or SLB occurring during this period. The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident

BASES

ACTIONS

A.1 (continued)

occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one or both of the containment cooling fan coil units (FCU) in one train inoperable, the inoperable FCU(s) must be restored to OPERABLE status within 7 days. In this degraded condition the remaining OPERABLE containment spray and cooling trains provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

BASES

ACTIONS

C.1 (continued)

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1 and D.2

Condition D applies when one FCU in each train is inoperable. With two FCUs inoperable, the Required Actions are to isolate cooling water flow to both inoperable FCUs immediately. This will assure the containment cooling function continues to be provided.

The LCO requires the OPERABILITY of a number of components within the subsystems. Due to the redundancy of components within the containment cooling system, the inoperability of two FCU does not render the containment cooling system incapable of performing its function. Engineering analyses demonstrate that two OPERABLE FCUs, one in each train, are capable of providing the necessary cooling.

With a FCU inoperable in both containment cooling trains and a FCU OPERABLE in both containment cooling trains, the two remaining OPERABLE FCUs can provide the necessary cooling provided the cooling water flow to the inoperable FCUs is isolated.

When one FCU in each containment cooling train is inoperable, both inoperable FCUs must be restored to OPERABLE status within 7 days. In this degraded condition the remaining OPERABLE containment spray and FCU from each cooling train provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the heat removal capabilities afforded by combinations of the Containment Spray System and Containment

BASES

ACTIONS

D.1 and D.2 (continued)

Cooling System and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action D.2 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.2

Operating each containment fan coil unit on low motor speed for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly.

Motor current is measured and compared to the nominal current expected for the test condition. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan coil units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between Surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that cooling water flow rate to each containment fan coil unit is ≥ 900 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4).

Terminal temperatures of each fan coil unit are also observed. This test includes verifying operation of all essential features including low motor speed, cooling water valves and normal ventilation system dampers. The 24 month Frequency is based on; the need to perform these Surveillances under the conditions that apply during a plant outage; the known reliability of the Cooling Water System; the two train redundancy available; and, the low probability of a significant degradation of flow occurring between Surveillances.

SR 3.6.5.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code. Since the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.4 (continued)

containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-High pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. To prevent inadvertent spray in containment, containment spray pump testing with a simulated actuation signal will be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. These tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.5.8

With the spray header drained, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 37, 38, 41, 42, 49, 52, and 58 through 61 issued for comment July 10, 1967, as referenced in USAR Section 1.2.
2. USAR Section 6.4.
3. USAR, Section 14.5.
4. USAR, Section 6.3.
5. USAR, Section 5.2.

BASES

BACKGROUND (continued)

a. One motor driven AFW pump;

b. One turbine driven AFW pump;

c. Steam generator AFW motor-operated supply valves; and

d. Steam generator AFW motor-operated throttle valves.

These components are configured to provide a flow path from each pump to both steam generators for the specific unit.

Each motor driven or turbine driven AFW pump can provide 100% of the required AFW flow capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

The turbine driven AFW pump receives steam from both main steam lines upstream of the main steam isolation valves. Each steam feed line will supply 100% of the requirements of the turbine driven AFW pump. An air operated valve downstream of the motor operated valves from each loop allows passage of steam to the turbine driven AFW pump when required. The air supply to the valve is controlled by a normally open DC solenoid valve designed such that failure of either the air supply or control power would cause the respective valve to open, starting the turbine driven AFW pump. Additionally, the air operated steam supply valve has a safety function to close on turbine driven AFW pump low suction or discharge pressure, which results in tripping the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit operation in MODES 2 and 3. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions.

The AFW System is designed to supply sufficient water to the steam

BASES

BACKGROUND (continued)

generators to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the SG PORVs or steam dump valve.

The following safety signals automatically initiate an AFW pump start signal:

a. Low-low water level in either steam generator; and

b. Safety injection.

Additionally, the following signals initiate an AFW pump start signal:

a. Trip of both main feedwater pumps (bypassed during startup and shutdown operation);

b. Loss of both 4 kV normal buses (turbine driven AFW pump only); and

c. Manually either local or remote.

Depending on pump type, the motor will start or the turbine steam admission air operated control valve will open.

The AFW System is discussed in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event involving loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus margin for uncertainty and accumulation.

BASES

LCO (continued)

c. MV-32034 or MV-32035 are closed and the associated breaker is locked in the OFF position;

d. MV-32036 and MV-32037 are open and both breakers are locked in the OFF position; and

e. Bus 27 is supplied from Bus 26.

Changes in valve positions to align 121 CL pump as the safeguards substitute for either diesel driven CL pump must be under direct administrative control.

With one CL strainer isolated, the containment cooling train on the associated CL header is OPERABLE at CL supply temperatures up to and including 70°F. When the CL supply temperature is above 70°F with one CL strainer isolated, the containment cooling train on the associated CL header is not OPERABLE. If Technical Specification (TS) 3.6.5 Condition D has been entered, then the above correlation between CL strainer status, CL supply temperature and containment cooling train OPERBILITY is not applicable. In this case the remaining two containment cooling fan coil units provide adequate heat removal within the TS 3.6.5 Condition D allowed Completion Time.

A header is considered to be OPERABLE when the associated piping, valves, and instrumentation and controls can perform the required safety related functions:

a. Provide flow and cooling for the required safeguards components supplied from the header; and

b. Provide necessary isolation functions required for the header during a safeguards actuation.

Removal of return header piping or components from service does not automatically make the system inoperable. Factors to consider during an OPERABILITY determination are:

BASES

LCO (continued)

a. If the piping or component inoperability results in an individual component being incapable of heat removal, the individual component is to be considered inoperable;

b. If the piping or component inoperability results in required components in a train being incapable of heat removal, the train is to be considered inoperable; and

c. If cooling flow for the required components can be maintained by opening the emergency dump to grade path, by routing to the other unit's discharge header, or overflow from the turbine building standpipes, the train or components are not considered inoperable.

APPLICABILITY

The CL System specification is applicable for single or two unit operation.

In MODES 1, 2, 3, and 4, the CL System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the CL System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the CL System are determined by the systems it supports.

ACTIONS

A.1

If no safeguards CL pumps are OPERABLE for one train, action must be taken to restore one CL safeguards pump to OPERABLE status within 7 days.

Either the diesel driven CL pump for the train may be restored to OPERABLE status, or the 121 CL pump may be aligned to fulfill the

BASES

ACTIONS

A.1 (continued)

safeguards function for the train that has no OPERABLE safeguards CL pump.

The 7 day Completion Time is based on:

a. Low probability of loss of offsite power during the period;

b. The low probability of a DBA occurring during this time period;

c. The safeguards cooling capabilities afforded by the remaining OPERABLE train; and

d. The capability to route water from the non-safeguards pumps, if needed.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for combinations of Conditions A and B to be inoperable during any continuous failure to meet this LCO for these Conditions.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

Required Action A.1 is modified by 3 Notes. Note 1 requires Unit 1 entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," for an emergency diesel generator made inoperable by the CL System. For Unit 1, the diesel generators are major heat loads supplied by the CL System. Thus, inoperability of two safeguards CL pumps will affect at least the heat

BASES

ACTIONS

A.1 (continued)

loads on one CL header, including one Unit 1 diesel generator. Inability to adequately remove the heat from the diesel generator will render it inoperable.

Note 2 requires entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4", for both units for the RHR loops made inoperable by the CL System. If either unit is in MODE 4, inoperability of two safeguards CL pumps may affect all the heat loads on one CL header, including a CC train and subsequently one RHR heat exchanger on each unit. Inability to adequately remove the heat from a RHR heat exchanger will render it inoperable.

Note 3 specifies that the Condition with no safeguard CL pumps OPERABLE for one train may not exist for more than 7 days in any consecutive 30 day period. If such a condition occurs, Condition C must be entered with the specified Required Action taken because the equipment reliability is less than considered acceptable.

B.1, B.2 and B.3

If one CL supply header is inoperable, action must be taken to verify the vertical motor driven CL pump and the opposite train diesel driven CL pump are OPERABLE within 4 hours, and restore the inoperable CL header to OPERABLE status within 72 hours.

Verification of vertical motor driven CL pump OPERABILITY does not require the pump to be aligned and may be performed by administrative means. Verification of the opposite train diesel driven CL pump may be performed by administrative means. Completion of the CL pump surveillance tests is not required.

Conditions may occur in the CL System piping, valves, or instrumentation downstream of the supply header (e.g., closed or

BASES

ACTIONS

B.1, B.2 and B.3 (continued)

failed valves, failed piping, or instrumentation in a return header) that can result in the supply header being considered inoperable. In such cases, Condition B and related Required Actions shall apply.

In this Condition, the remaining OPERABLE CL header is adequate to perform the heat removal function. However, the overall redundancy is reduced because only a single CL train remains OPERABLE.

Required Action B.1 ensures that the vertical motor driven 121 CL pump may be used to provide redundancy for the safeguards CL pump on the OPERABLE header. Required Action B.3 assures adequate system reliability is maintained.

The second Completion Time for Required Action B.3 establishes a limit on the maximum time allowed for combinations of Conditions A and B to be inoperable during any continuous failure to meet this LCO for these Conditions.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

Required Actions B.1, B.2, and B.3 are modified by two Notes.

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," should be entered for Unit 1 since an inoperable CL train results in an inoperable emergency diesel generator.

BASES

ACTIONS

B.1, B.2 and B.3 (continued)

The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," should be entered if an inoperable CL train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

The 4 and 72 hour Completion Times are based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period. In addition, the 4 hour Completion Time for Required Actions B.1 and B.2 is within the time period anticipated to verify OPERABILITY of the required CL pump by administrative means.

C.1 and C.2

If at least one safeguards CL pump for a train or a CL supply header cannot be restored to OPERABLE status within the associated Completion Time, the units must be placed in a MODE in which the LCO does not apply. To achieve this status the units must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

In this Condition, the 14 day fuel oil supply for the diesel driven CL pumps is not available. However, the Condition is restricted to fuel oil supply reductions that maintain at least a 12 day supply. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank(s). A period of 48 hours is considered sufficient to complete restoration of the required supply prior to

BASES

ACTIONS

D.1 (continued)

declaring the diesel driven CL pumps inoperable. This period is acceptable based on the remaining 12 day fuel oil supply, the fact that procedures will be initiated to obtain replenishment, availability of the vertical motor driven CL pump and the low probability of an event during this brief period.

The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for combinations of Conditions A and D to be inoperable during any continuous failure to meet this LCO for these Conditions.

The 9 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and D are entered concurrently. The AND connector between 48 hours and 9 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

E.1

With the stored fuel oil supply not within the limits specified or Required Actions and associated Completion Times of Condition D not met, the diesel driven CL pumps may be incapable of performing their intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the CL System components or systems may render those components inoperable, but does not affect the OPERABILITY of the CL System.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool Storage

BASES

BACKGROUND

The spent fuel storage pool is a two compartment pool as described in the USAR (Ref. 1). These 2 compartments are referred to as Pool 1 and Pool 2.

Criticality considerations provide the primary basis for storage limitations.

Pool 1 may contain up to 462 storage positions, except when the pool is used for cask laydown. In the latter case, only 266 storage positions are available since 4 storage racks must be removed to accommodate the storage cask. Pool 2 has up to 1120 storage positions.

Pools 1 and 2 are designed to accommodate fuel of various initial enrichments (up to 5 weight percent (w/o)), which have accumulated minimum burnups and decay times within the unrestricted domain according to Figure 3.7.17-1 in the accompanying LCO.

Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 1.00 be evaluated in the absence of soluble boron. The double contingency principle discussed in Reference 2 and the April 1978 NRC letter (Ref. 3) allows credit for additional soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel pool may therefore be achieved by controlling the location of each

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BACKGROUND (continued)	assembly in accordance with the accompanying LCO and maintaining boron concentration in accordance with LCO 3.7.16.

APPLICABLE SAFETY ANALYSES

The hypothetical criticality accidents can only take place during or as a result of the movement of an assembly (Ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Fuel Storage Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by verifying the appropriate checkerboarding after each fuel handling campaign, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for criticality accidents, the operation may be under the auspices of the accompanying LCO.

The spent fuel storage racks have been analyzed in accordance with the methodology contained in Reference 4. That methodology ensures that the spent fuel rack multiplication factor, k_{eff}, is less than 0.95 as recommended by ANSI 57.2-1983 (Ref. 6) and NRC guidance (Ref. 3). The codes, methods and techniques contained in the methodology are used to satisfy this criterion on k_{eff}. The resulting Prairie Island spent fuel rack criticality analysis allows for the storage of fuel assemblies with enrichments up to a maximum of 5.0 (nominal 4.95% $\pm$ 0.05%) weight percent U-235 while maintaining $k_{eff} \leq 0.95$ including uncertainties and credit for soluble boron. In addition, sub-criticality of the pool ($k_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. Credit is taken for radioactive decay time of the spent fuel and for the presence of fuel rods containing gadolinium burnable poison.

The criticality analysis (Ref. 4) utilized the following storage configurations to ensure that the spent fuel pool will remain subcritical during the storage of fuel assemblies with all possible combinations of burnup and initial enrichment:

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APPLICABLE SAFETY ANALYSES (continued)

a. The first storage configuration utilizes a pattern to accommodate new or low burnup fuel with maximum enrichment of 5.0 w/o U-235. This configuration stores "burned" and "fresh" fuel assemblies in a 3x3 checkerboard pattern as shown in Figure 4.3.1-1. Fuel assemblies stored in burned" cell locations are selected based on a combination of initial enrichment, discharge burnup and decay time (Figures 4.3.1-3 and 4.3.1-4). The criteria for the fuel stored in the "burned" locations is also dependent on the presence of rods containing gadolinium in the center "fresh" fuel assembly. The use of empty cells is also an acceptable option for the "fresh" and "burned" cell locations. This will allow the storage of new or low burnup fuel assemblies in the outer rows of the spent fuel storage racks because the area outside the racks can be considered to be empty cells.

   Fuel assemblies that fall into the restricted range of Figure 3.7.17-1 are required to be stored in "fresh" cell locations as shown in Figure 4.3.1-1. The criteria included in Figure 3.7.17-1 for the selection of fuel assemblies to be stored in the "fresh" cell locations is based on a combination of initial enrichment, decay time and discharge burnup.

b. The second storage configuration does not utilize any special loading pattern. Fuel assemblies with burnup, initial enrichment and decay time which fall into the unrestricted range of Figure 3.7.17-1 can be stored anywhere in the region with no special placement restrictions.

The burned/fresh fuel checkerboard region can be positioned anywhere within the spent fuel racks, but the boundary between the checkerboard region and the unrestricted region must be either:

a. Separated by a vacant row of cells; or

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APPLICABLE SAFETY ANALYSES (continued)

b. The interface must be configured such that there is one row carryover of the pattern of burned assemblies from the checkerboard region into the first row of the unrestricted region (Figure 4.3.1-2).

Specification 3.7.17 and Section 4.3 ensure that fuel is stored in the spent fuel racks in accordance with the storage configurations assumed in the spent fuel rack criticality analysis (Ref. 4).

The spent fuel pool criticality analysis addresses all the fuel types currently stored in the spent fuel pool and in use in the reactor. The fuel types considered in the analysis include the Westinghouse Standard (STD), OFA, and Vantage Plus designs, and the Exxon fuel assembly types in storage in the spent fuel pool.

Accident conditions which could increase the k_{eff} were evaluated including:

a. A new fuel assembly drop on the top of the racks;

b. A new fuel assembly misloaded between rack modules;

c. A new fuel assembly misloaded into an incorrect storage rack location;

d. Intramodule water gap reduction due to a seismic event; and

e. Spent fuel pool temperature greater than 150°F.

For an occurrence of these postulated accident conditions, the double contingency principle of Reference 2 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 464 ppm required to maintain k_{eff} less than 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its

BASES

APPLICABLE SAFETY ANALYSES (continued)

presence would be a second unlikely event.

Westinghouse Electric Company LLC calculations (Ref. 4) were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by these postulated accidents and to maintain k_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 730 ppm was adequate to mitigate these postulated criticality related accidents and to maintain k_{eff} less than or equal to 0.95.

Specification 3.7.16 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.7.16 is consistent with the boron concentration limit required for a spent fuel cask containing fuel.
a mispositioned fuel assembly or a loss of spent fuel pool cooling. The 1800 ppm spent fuel pool boron concentration limit in Specification 3.7.16 is consistent with the boron concentration limit required for a spent fuel cask containing fuel.

Section 4.3 requires that the spent fuel rack k_{eff} be less than or equal to 0.95 when flooded with water borated to 750 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not a credible event.

When the requirements of Specification 3.7.17 are not met, immediate action must be taken to move any noncomplying fuel assembly to an acceptable location to preserve the double contingency principle assumption of the criticality accident analysis.

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APPLICABLE SAFETY ANALYSES (continued)	The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.17-1 in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.95, with credit given for boron in the water. Fuel assemblies not meeting the criteria of Figure 3.7.17-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.
ACTIONS	<u>A.1</u> Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in the spent fuel storage pool is not in accordance with Figure 3.7.17-1 or Specification 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.17-1 or Specification 4.3.1.1. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 in the accompanying LCO. For fuel assemblies in the restricted range of Figure 3.7.17-1 performance of this SR will ensure compliance with Specification 4.3.1.1.

The Frequency of this SR is prior to storing or moving a fuel assembly.

SR 3.7.17.2

This SR verifies that the fuel assemblies in the spent fuel storage racks are stored in accordance with the requirements of LCO 3.7.17 and Section 4.3.1.1.

The intent of this SR is to not require completion of the spent fuel pool inventory verification during interruptions in fuel handling during a defined fuel handling campaign. No spent fuel pool inventory verification is required following fuel movements where no fuel assemblies are relocated to different spent fuel rack locations.

The Frequency of this SR requires performance within 7 days after the completion of any fuel handling campaign which involves:

a. The relocation of fuel assemblies within the spent fuel pool; or

b. The addition of fuel assemblies to the spent fuel pool.

The extent of a fuel handling campaign will be defined by plant administrative procedures. Examples of a fuel handling campaign would include all the fuel handling performed during a refueling outage or associated with the placement of new fuel into the spent fuel pool.

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SURVEILLANCE REQUIREMENTS

SR 3.7.17.2 (continued)

The 7 day allowance for completion of this SR provides adequate time for completion of the spent fuel pool inventory verification while minimizing the time a fuel assembly may be misloaded in the spent fuel pool. If a fuel assembly is misloaded during the fuel handling campaign, the minimum boron concentration required by LCO 3.7.16 will ensure that the spent fuel rack k_{eff} remains within limits until the spent fuel inventory verification is performed

REFERENCES

1. USAR, Section 10.2.

2. ANSI/ANS-8.1-1983.

3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978.

4. "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis", Westinghouse Electric Company calculation CN WFE 03-40, November 11, 2004.

5. Not Used.

6. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants", ANSI/ANS-57.2-1983, October 7, 1983.