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

Technical Specifications Bases Revisions

Nuclear Management Company, LLC (NMC), licensee for the Point Beach Nuclear Plant (PBNP), Units 1 and 2, hereby submits a revision to the Technical Specifications (TS) Bases for the following TSs: LCO 3.1.1, "Shutdown Margin (SDM)", LCO 3.1.4, "Rod Group Alignment Limits", LCO 3.3.5, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation", LCO 3.6.3, "Containment Isolation Valves", LCO 3.7.8, "Service Water (SW) System", LCO 3.8.1, "AC Sources – Operating", LCO 3.8.2, "AC Sources – Shutdown", LCO 3.8.6, "Battery Cell Parameters", and LCO 3.9.3, "Containment Penetrations" (two revisions). A description of the changes is provided in Enclosure 1.

These changes have been screened for evaluation pursuant to the requirements of 10 CFR 50.59 in accordance with approved PBNP procedures and were determined to be acceptable.

Enclosure 2 provides clean copies of the affected TS Bases pages indicating the changes. These are provided for your information in accordance with the TS Bases Control Program.

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Enclosures: 1 - Description of Changes
2 - Revised Technical Specifications Bases Pages

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ENCLOSURE 1
TECHNICAL SPECIFICATIONS BASES REVISIONS
DESCRIPTION OF CHANGES

1.0 INTRODUCTION

Nuclear Management Company, LLC (NMC), licensee for the Point Beach Nuclear Plant (PBNP), Units 1 and 2, hereby submits a revision to the following Bases for Technical Specifications (TS):

- LCO 3.1.1, "Shutdown Margin (SDM)",
- LCO 3.1.4, "Rod Group Alignment Limits",
- LCO 3.3.5, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation",
- LCO 3.6.3, "Containment Isolation Valves",
- LCO 3.7.8, "Service Water (SW) System",
- LCO 3.8.1, "AC Sources – Operating",
- LCO 3.8.2, "AC Sources – Shutdown",
- LCO 3.8.6, "Battery Cell Parameters", and
- LCO 3.9.3, "Containment Penetrations" (two revisions).

2.0 DESCRIPTION OF CHANGES

LCO 3.1.1, "Shutdown Margin (SDM)"

The Bases for LCO 3.1.1 were revised to add a discussion of the most limiting main steam line break accident, as stated in Westinghouse letter NSAL 02-014, Steam Line Break During Mode 3.

LCO 3.1.4, "Rod Group Alignment Limits"

The Bases for LCO 3.1.4 were revised to reflect License Amendments 212 and 217 for Units 1 and 2, respectively, which were issued March 29, 2004.

LCO 3.3.5, "Control Room Emergency Filtration System (CREFS) Actuation Instrumentation"

The Bases for LCO 3.3.5 were revised to delete reference to the containment isolation signal, since this signal is not part of the CREFS actuation instrumentation.

LCO 3.6.3, "Containment Isolation Valves"

The Bases for LCO 3.6.3 were revised to reflect a modification to the Unit 1 containment purge valves. The Unit 1 containment purge supply and exhaust outboard valves have been removed. A corresponding modification was performed on Unit 2 during its previous refueling outage. Blind flanges replace the outboard valves to provide containment isolation during MODES 1, 2, 3, and 4. The inboard valves remain in place as containment isolation valves.

LCO 3.7.8, "Service Water (SW) System"

The Bases for LCO 3.7.8 were revised to update References 5 and 6 to the current PBNP SW calculation (PBNP Calculation 2002-0003, Service Water System Design Basis).

LCO 3.8.1, "AC Sources – Operating"

The Bases for LCO 3.8.1 were revised to clarify the discussion regarding which feeder and bus tie breakers on the 480V safeguards bus trip in response to a safety injection signal coincident with an undervoltage condition. Reference to the auxiliary feedwater pump motor was deleted.

LCO 3.8.2, "AC Sources – Shutdown"

The Bases for LCO 3.8.2 were revised to clarify that an operable standby emergency power source (i.e., diesel generator) is required for only one of the distribution system train(s) required to be operable by LCO 3.8.10. A reference to the NRC Policy Statement was updated to refer to 10 CFR 50.36(c)(2)(ii).

LCO 3.8.6, "Battery Cell Parameters"

The Bases for LCO 3.8.6 were revised to correct a typographical error in Reference 2

LCO 3.9.3, "Containment Penetrations" (two revisions)

The Bases for LCO 3.9.3 were revised to clarify administrative controls for implementing closure when maintaining airlock doors open. This change reflects the requirements of License Amendments 213 and 218 for Units 1 and 2 respectively, which were issued April 2, 2004.

A subsequent revision to the Bases for LCO 3.9.3 was made to reflect a modification to the Unit 1 containment purge valves. The Unit 1 containment purge and exhaust outboard valves have been removed. A corresponding modification was performed on Unit 2 during its previous refueling outage. Blind flanges replace the outboard valves to provide containment isolation during MODES 1, 2, 3, and 4. The valves are reinstalled prior to entry into MODE 6. During MODES 1, 2, 3, and 4, each of the purge supply and exhaust isolation valves are secured in the closed position. An additional change consisted of a reference to the NRC Policy Statement being updated to refer to 10 CFR 50.36(c)(2)(ii).

ENCLOSURE 2
TECHNICAL SPECIFICATION BASES REVISIONS

Affected TS Bases Pages:

B 3.1.1-1 through B 3.1.1-6
B 3.1.4-2 through B 3.1.4-10
B 3.3.5-1
B 3.6.3-1
B 3.7.8-2, B 3.7.8-3, and B 3.7.8-9
B 3.8.1-3 and B 3.8.1-4
B 3.8.2-2
B 3.8.6-6
B 3.9.3-3 (first revision)
B 3.9.3-2 (second revision)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to the Point Beach Design Criteria (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks within the limits of LCO 3.1.5, "Shutdown Bank Insertion Limits," and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Refs. 2 and 5) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth

BASES

APPLICABLE SAFETY ANALYSES (continued)

rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 225 cal/gm energy deposition for unirradiated and ≤ 200 cal/gm energy deposition for irradiated fuel during a rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

The analysis has demonstrated that the most limiting MSLB accident is from an initial condition of hot zero power. Westinghouse has concluded that this analyzed condition (Ref. 6) remains bounding for operating MODES below MODE 1, hot zero power, given that:

a. The safety injection system is still available to provide boration; or,

BASES

APPLICABLE SAFETY ANALYSES (continued)	b. The RCS is borated to cold shutdown boron concentration prior to the safety injection system block setpoint being reached during Mode 3.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition; and

c. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

The uncontrolled rod withdrawal transient is terminated by a high power level trip or an OT ΔT trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

BASES

LCO (continued)

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM limit. For MSLB accidents, if the limit is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the limit is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with $k_{eff} \geq 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 32 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 32 gpm and 3.75% boric acid represent typical values and are provided for the purpose of offering a specific example.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $k_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth. In MODE 6, SDM is verified by observing that the requirements of LCO 3.9.1, "Boron Concentration" are met.

In MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5, the SDM is verified, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control and shutdown bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

BASES

REFERENCES

1. FSAR, Section 3.1.
2. FSAR, Section 14.2.5.
3. FSAR, Section 14.1.4.
4. 10 CFR 100.
5. FSAR, Sections 14.1.1 and 14.2.6.
6. Westinghouse NSAL 02-014, Steam Line Break During Mode 3.

BASES

BACKGROUND (continued)

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Rod Position Indication (RPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides a highly accurate indication of actual rod position, but at a lower precision than the step counters. The RPI is a linear variable differential transformer (LVDT) consisting of primary and secondary coils stacked alternately on a support tube with the control rod drive shaft acting as the core of the transformer. The primary and secondary coils are series connected with the primary coil supplied with AC power from a constant current source. The position of the control rod drive shaft changes the primary to secondary coil magnetic coupling resulting in a variable secondary voltage which is proportional to the position of the drive shaft (control rod). The RPI channel has an indication accuracy of 5% of span (11.5 steps) therefore, the maximum deviation between actual and demanded indication could be 36 steps (24 steps maximum allowable deviation plus 12 steps indication accuracy).

The specifications ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of rod misalignment on associated accident

BASES

BACKGROUND (continued)

analyses are limited. Operability of the control rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits.

Permitted control rod misalignments (as indicated by the RPI System within one hour after control rod motion) are; a) ± 18 steps of the bank demand position (if sufficient peaking factor margin exists, the power level is greater than 85 percent of rated power, and bank D demand is less than 215 steps withdrawn), b) ± 24 steps of the bank demand position (if sufficient peaking factor margin exists, the power level is greater than 85 percent of rated power, and bank D demand is greater than 215 steps withdrawn), and c) ± 24 steps of the bank demand position (if the power level is less than or equal to 85 percent of rated power). Above 85 percent of rated power, sufficient peaking factor margin is demonstrated by satisfying the requirements of Table 3.1.4-1, e.g., for an 18 step indicated misalignment and rods less than 215 steps withdrawn, the peak measured $F_Q(Z)$ from the latest incore flux map must be at least 5.0% less than the limit and the peak measured $F^N_{\Delta H}$ from the latest incore flux map must be at least 2.0% less than the limiting value. For power levels less than or equal to 85 percent of rated power, the peaking factor margin does not have to be verified on an explicit basis. This is due to the rate of peaking factor margin increase (due to the peaking factor limit increasing) as the power level decreases being greater than the peaking factor margin loss (due to the increased control rod misalignment). This effect is described in WCAP-15432 Rev. 2. These limits are applicable to all shutdown and control rods (of all banks) over the range of 0 to 230 steps withdrawn inclusive.

Control rods in a single bank move together with no individual rod insertion differing by more than 30 steps from the bank demand position (operation at greater than 85 percent of rated power and demand less than 215 steps), nor more than 36 steps (operation at less than or equal to 85 percent of rated power or operation at greater than 85 percent of rated power and demand position greater than or equal to 215 steps withdrawn). An indicated misalignment limit of 18 steps precludes a rod misalignment of greater than 30 steps with consideration of instrumentation error; 24 steps indicated misalignment corresponds to 36 steps with instrumentation error.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 4). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:
   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 36 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factors $F_Q^C(Z)$ and $F_Q^W(Z)$ and the nuclear enthalpy hot channel factor $F_{\Delta H}^N$ are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q^C(Z)$, $F_Q^W(Z)$ and $F_{\Delta H}^N$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q^C(Z)$, $F_Q^W(Z)$, and $F_{\Delta H}^N$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

BASES

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirement is satisfied provided the control rod will fully insert within the required rod drop time assumed in the safety analysis. Control rod malfunctions that result in the inability to move a control rod (e.g. lift coil and rod control system logic failures), but do not impact the control rod trippability, do not result in control rod inoperability. The LCO requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within ± 18 steps (for power operation above 85% and bank demand position less than 215 steps) or within ± 24 steps (for power operation greater than 85% and bank demand position greater than or equal to 215 steps) is conservative. The minimum misalignment assumed in safety analysis is 36 steps, and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" for SDM in MODE 2 with $k_{eff} < 1.0$, and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

BASES

ACTIONS

The ACTIONS table is modified by a Note indicating that verification of rod operability and the comparison of bank demand position and RPI System may take place at any time up to one hour after rod motion, at any power level. This allows up to one hour of thermal soak time to allow the control rod drive shaft to reach a thermal equilibrium and thus present a consistent position indication. For purposes of invoking this allowance, a substantial rod movement is required. Substantial rod movement is considered to be 10 or more steps in one direction in less than or equal to one hour.

A.1.1 and A.1.2

When one or more rods are inoperable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion

Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM. In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

ACTIONS (continued) B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 25 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors $F_Q^C(Z)$, $F_Q^W(Z)$ and $F_{\Delta H}^N$ must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 4). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q^C(Z)$, $F_Q^W(Z)$ and $F_{\Delta H}^N$, are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q^C(Z)$, $F_Q^W(Z)$ and $F_{\Delta H}^N$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses

ACTIONS (continued) presented in the FSAR Chapter 14 (Ref. 4) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position.

The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

BASES

REFERENCES

1. FSAR, Section 3.2.
2. FSAR, Sections 1.3.5.
3. 10 CFR 50.46.
4. FSAR, Chapter 14.

B 3.3 INSTRUMENTATION

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

BASES

BACKGROUND

The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. The control room ventilation system normally operates in the normal operating mode (Mode 1). Upon receipt of an actuation signal, the CREFS initiates the emergency make-up (Mode 4) mode of operation. The control room ventilation system and its operating modes are described in the Bases for LCO 3.7.9, "Control Room Emergency Filtration System."

The actuation instrumentation consists of noble gas radiation monitor in the air intake and control room area radiation monitor. A high radiation signal from either of these detectors will initiate the emergency make-up mode of operation (Mode 4) of the CREFS.

APPLICABLE SAFETY ANALYSES

The CREFS provides airborne radiological protection for control room personnel, as demonstrated by the limiting control room dose analyses for the design basis large break loss of coolant accident. Control room dose analysis assumptions are presented in the FSAR, Section 14.3.5 (Ref. 1).

In MODES 1, 2, 3, and 4, the CREFS radiation monitor actuation signal will provide automatic initiation of CREFS in the emergency make-up mode of operation (Mode 4) during design basis events which result in significant radiological releases to the environs (e.g. large break loss of coolant accident, steam generator tube rupture, reactor coolant pump locked rotor, etc;).

The CREFS radiation monitor actuation signal also provides automatic initiation of CREFS, in the emergency make-up mode of operation (Mode 4), to assure control room habitability in the event of a fuel handling during movement of
irradiated fuel, and CORE ALTERATIONS.

Further Applicable Safety Analysis information for CREFS is contained in the Bases for LCO 3.7.9, "Control Room Emergency Filtration System."

The CREFS actuation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for penetrations to be provided with two isolation barriers. These isolation barriers are either passive or active. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive barriers. Valves designed to close either automatically or manually (including check valves with flow through the valve not secured), are considered active barriers. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active barrier can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. These barriers (typically containment isolation valves) make up the Containment Isolation System.

An automatic containment isolation signal is produced upon receipt of a safety injection signal. The containment isolation signal isolates process lines in order to minimize leakage of fission product radioactivity. As a result, the containment isolation valves (and passive barriers) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Loss of Coolant Accident (LOCA).

The OPERABILITY requirements for containment isolation valves help ensure that containment integrity is established and maintained in accordance with the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

<u>Containment Purge System (purge supply and exhaust valves)</u>

The Containment Purge System can be operated to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment whenever the unit is not in MODES 1, 2, 3, or 4. The purge supply and exhaust lines each contain one isolation valve. Blind flanges replace the purge supply and exhaust outboard valves to provide containment isolation during MODES 1, 2, 3, and 4. The valves are reinstalled prior to entry into MODE 6. The inboard valves remain in place as containment isolation valves.

BASES

BACKGROUND (continued)	Emergency Diesel Generator output breaker closure generates an automatic sequenced start of the SW pumps in order to meet the immediate cooling needs of G-01 and/or G-02 upon a Loss of Offsite Power event. This function is not required for SW pump OPERABILITY. The Bases for LCO 3.8.1, "AC Sources – Operating," provides information regarding this function on standby emergency power OPERABILITY. Additional information about the design and operation of the SW System, along with a list of the components served, is presented in the FSAR, Section 9.6 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The design basis of the SW System is three SW pumps when both units are in MODES 1, 2, 3 or 4 (Ref. 5), in conjunction with the CCW System and a 100% capacity containment cooling system, to remove core decay heat following a design basis LOCA as discussed in the FSAR, Section 14.3.4 (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ECCS pumps. The SW System is designed to perform its function with a single failure of any active component, assuming the loss of offsite power. The SW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in the FSAR, Section 9.2, (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR System pumps and heat exchangers that are operating. Heat transferred from the reactor core to the SW System during accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation is removed by Lake Michigan. Operating limits for the SW System are based on the approved SW System analyses as stated in Appendix C, Additional Conditions, Operating Licenses DPR-24 and DPR-27. The SW System satisfies Criterion 3 of the NRC Policy Statement.
LCO	The SW System is required to be OPERABLE to provide the required redundancy to ensure that the system will function to remove post accident heat loads, assuming the worst case single active failure. The SW System is OPERABLE during MODES 1, 2, 3, and 4 when: a. six SW pumps are OPERABLE;

BASES

LCO (continued)

b. the SW ring header continuous flowpath is not interrupted;

c. the required non-essential-SW-load isolation valves are OPERABLE or the affected non-essential flowpath is isolated;

d. the opposite unit's containment fan cooler SW outlet motor operated valves are closed or the SW flowpath is isolated; and

e. the instrumentation and controls required to perform the safety related function are OPERABLE.

The LCO is modified by a Note indicating that only five SW pumps are required to be OPERABLE with one unit in MODE 5 or 6, or defueled, and the SW System is capable of providing required cooling water flow to required equipment. Operation of the SW System with five operable Service Water pumps for an indefinite period of time is allowed if the system is in a configuration that ensures that all relevant design basis requirements are met while sustaining the most limiting single active failure. If only five SW pumps are operable, the most limiting single active failure is the loss of the safeguards train that provides the "Start" signal for three of the operable SW pumps.

Calculations (Ref. 5) demonstrate that two operating SW pumps are sufficient to meet all the current design basis acceptance criteria, provided that enough SW flowpaths are isolated prior to the postulated accident. The required SW configuration is specified in TRM 3.7.7. The isolation of these flowpaths necessitates that the non-accident unit be in MODE 5 or 6, or defueled.

APPLICABILITY

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

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

ACTIONS

The Actions Table is modified by a Note which requires the applicable Conditions and Required Actions to be entered for the system made inoperable as a result of any SW System inoperability. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the SW System non-essential-SW-load isolation valves on an actual or simulated actuation signal. The SW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the SW System pumps on an actual or simulated actuation signal. The SW System is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR. Section 9.6.
2. FSAR. Section 14.3.4.
3. FSAR. Section 9.2.
4. Technical Requirements Manual, TLCO 3.7.7, SW System
5. PBNP Calculation 2002-0003, Service Water System Design Basis

BASES

BACKGROUND (continued)

initiation signal(s) which are received. The standby emergency power sources will start and operate in the standby mode without tying to its respective 4.16 kV ESF bus(es) on an SI signal alone.

Response to a undervoltage condition alone is as follows:

a. The standby emergency power source(s) auto starts.

b. Trip of the 4.16 kV bus supply breaker(s).

c. All feeder and bus tie breakers on the 480 V safeguards bus, except for the component cooling pump motor, auxiliary feedwater pump motor, and the safeguards motor control centers, are tripped. For the A train 480 V buses, this load shedding function is blocked after the bus emergency diesel generator output circuit breaker closes. This is necessary to prevent inadvertent load shedding during load sequencing. For the train B buses, this load shedding function is not blocked. The train B emergency diesel generator transient voltage response is sufficient to maintain bus voltage above the 480 VAC Loss of Voltage Relay setpoint during load sequencing.

d. After the standby emergency power source comes up to speed (as sensed by diesel generator speed switches) and voltage (as determined by generator field being present), the associated standby emergency power source breaker closes, re-energizing the safeguards buses.

e. Automatic sequencing of SW pumps upon standby emergency power source breaker closure.

f. Manually start any auxiliary as required for safe plant operation.

Response to a Safety Injection signal, coincident with an undervoltage condition, is as follows:

a. The standby emergency power source(s) auto starts.

b. Trip of the 4.16 kV bus supply breaker(s) in response to the undervoltage condition.

c. All feeder and bus tie breakers on the 480 V safeguards bus, except for the safeguards motor control centers, are tripped. For the A train 480 V buses, this load shedding function is blocked after the bus emergency diesel generator output circuit breaker closes. This is necessary to prevent inadvertent load shedding during load sequencing. For the train B buses, this load shedding function

BASES

BACKGROUND (continued)

is not blocked. The train B emergency diesel generator transient voltage response is sufficient to maintain bus voltage above the 480 VAC Loss of Voltage Relay setpoint during load sequencing.

d. Automatic start of the component cooling pump motor is blocked, and the battery charger input contactors are tripped open.

e. After the standby emergency power source comes up to speed (as sensed by diesel generator speed switches) and voltage (as determined by generator field being present), the associated standby emergency power source breaker closes, re-energizing the safeguards buses.

f. Loading sequence of ESF equipment is initiated (refer to FSAR Section 8.8 for sequencer times).

g. Starting of containment spray pumps is independent of the ESF starting sequence. Containment spray start occurs within 10 seconds after a containment high pressure signal with the safeguards bus energized and may occur simultaneously with the start of other equipment.

The emergency generator automatic loading sequence, including engine starting, will be accomplished in approximately 60 seconds. The time between when the emergency diesel generator receives a start signal (i.e., after actuation of the 4.16 kV Loss of Voltage relay), until the emergency diesel generator is ready to accept load, shall not exceed 10 seconds.

The Train A standby emergency power sources (G01 and G02) are rated at 2,850 kW for 2000 hours, 0.8 power factor. Additional ratings for the Train A units include 2963 kW for 200 hours, 3000 kW for 4 hours and 3053 kW for a 30-minute period. The Train B standby emergency power sources are rated at 2848 kW for 2000 hours. Additional ratings for the Train B units include 2951 kW for 200 hours, and 2987 kW for 4 hours. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

The two Train A emergency diesel-generator sets are located in separate rooms in the seismic Class I section of the turbine building. The two Train B emergency diesel-generator sets are located in separate rooms in the seismic Class I Emergency Diesel Generator building.

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems-Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE standby emergency power source, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and standby emergency power source ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown.

The offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Offsite circuits are those that are described in the FSAR.

The AC electrical offsite sources for a unit in MODE 5 or 6 is described as follows:

One circuit between the offsite transmission network and the associated unit's 480 V Class 1E safeguards buses, B03 and B04, utilizing:

BASES

REFERENCES

1. FSAR. Chapter 14.

2. IEEE-450-1987.

BASES

BACKGROUND (continued)	The requirements for containment purge and exhaust system penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits. The Containment Purge and Exhaust System includes a 36 inch purge supply penetration and a 36 inch exhaust penetration. The purge supply and exhaust penetrations each contain one isolation valve. The purge supply and exhaust outboard valves have been removed. Blind flanges replace the outboard valves to provide containment isolation during MODES 1, 2, 3, and 4. The valves are reinstalled prior to entry into MODE 6. During MODES 1, 2, 3, and 4, each of the purge supply and exhaust isolation valves are secured in the closed position. The Containment Purge and Exhaust System is not subject to a Specification in MODE 5. In MODE 6, large air exchanges are necessary to conduct refueling operations. The 36 inch purge supply system is used for this purpose. All purge supply and exhaust valves are closed by the Containment Purge and Exhaust Isolation Instrumentation.
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, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 161 hours prior to movement of irradiated fuel 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. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). 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 any Containment Purge and Exhaust System penetration to be closed except for the OPERABLE containment purge and exhaust

BASES

LCO (continued)	penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure specified in the FSAR can be achieved. The containment personnel airlock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided that one door is capable of being closed in the event of a fuel handling accident. Administrative controls to implement closure will be in effect when maintaining airlock doors open. Hoses and cables running through an airlock shall employ a means to allow safe, quick disconnection or severance. No fuel movement may occur prior to the minimum decay time of 161 hours. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment. The allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident.
APPLICABILITY	The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment 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	<u>A.1 and A.2</u> If the containment equipment hatch, air locks, or any containment Purge and Exhaust System penetration is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.