T. PRESTON GILLESPIE, JR.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP / 7800 Rochester Hwy.
Seneca, SC 29672

864-873-4478
864-873-4208 fax
T.Gillespie@duke-energy.com

August 7, 2012

U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station
Docket Numbers 50-269, 50-270, and 50-287
Technical Specification Bases (TSB) Changes

Pursuant to Technical Specification 5.5.15, Technical Specifications (TS) Bases Control Program, please find attached the latest changes to the Oconee Technical Specification Bases (TSB). These changes remove obsolete information and provide the latest revisions of the affected Bases.

On July 13, 2012, Station Management approved a revision to TSB 3.1.4, Control Rod Group Alignment Limits; TSB 3.1.6, Axial Power Shaping Rod (APSR) Alignment Limits; and, TSB 3.1.7, Position Indicator Channels. The Bases were updated to remove information related to the analog Control Rod Drive System which has been replaced. Also included in this package are corrected pages for TSB 3.3.7, Engineered Safeguards Protective System (ESPS) Automatic Actuation Output Logic Channels and TSB 3.7.16, Control Room Area Cooling Systems (CRACS). These Bases were revised in response to issuance of Amendments 372, 274, and 373 which approved use of a Surveillance Frequency Control Program.

If any additional information is needed, please contact Kent Alter at 864-873-3255.

Sincerely,

TPGillespie

T. Preston Gillespie, Jr.
Vice President
Oconee Nuclear Station

A001
NRR

xc:

Mr. Victor M. McCree, Administrator, Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Mr. John P. Boska, Project Manager
(By electronic mail only)
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
One White Flint North, M/S O-8G9A
11555 Rockville Pike
Rockville, MD 20852

Mr. Andrew T. Sabisch
NRC Senior Resident Inspector
Oconee Nuclear Station

Ms. Susan E. Jenkins, Manager
Radioactive & Infectious Waste Management
SC Dept. of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

Proposed TS Bases Revision Remove/Insert Instructions

Remove Page	Insert Page
B 3.1.4-1	B 3.1.4-1
B 3.1.4-2	B 3.1.4-2
B 3.1.4-3	B 3.1.4-3
B 3.1.4-4	B 3.1.4-4
B 3.1.4-5	B 3.1.4-5
B 3.1.4-6	B 3.1.4-6
B 3.1.4-7	B 3.1.4-7
B 3.1.4-8	B 3.1.4-8
B 3.1.4-9	B 3.1.4-9
B 3.1.6-1	B 3.1.6-1
B 3.1.6-2	B 3.1.6-2
B 3.1.6-3	B 3.1.6-3
B 3.1.6-4	
B 3.1.7-1	B 3.1.7-1
B 3.1.7-2	B 3.1.7-2
B 3.1.7-3	B 3.1.7-3
B 3.1.7-4	B 3.1.7-4
B 3.3.7-7	B.3.3.7-7
	B 3.3.7-8
B 3.7.16-1	B 3.7.16-1
B 3.7.16-2	B 3.7.16-2
B 3.7.16-7	B 3.7.16-7

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of SDM. An inoperable CONTROL ROD that is unable to respond to positioning signals from the Rod Drive Control System may still meet its SDM capabilities if it is capable of responding to a valid trip signal (i.e., inoperable but trippable). It would, however, have the potential to adversely affect core power distribution due to its inability to maintain itself within the group average. An inoperable CONTROL ROD which is not "trippable" would satisfy neither the capacity to supply SDM requirements nor the ability to maintain itself in alignment with the group to assure acceptable core power distribution.

The applicable criteria for these design requirements are ONS Design Criteria (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CONTROL ROD to become inoperable or to become misaligned from its group. CONTROL ROD inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CONTROL RODS are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the roller nut assembly, but at different rates (jog and run) depending on the signal output from the Rod Drive Control System (RDCS).

BASES

BACKGROUND (continued)

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The CONTROL RODS provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity control during normal operation and transients, and their movement is normally governed by the Integrated Control System.

The axial position of CONTROL RODS is indicated by two separate and independent systems, which are the relative position indicator transducers and the absolute position indicator transducers (see LCO 3.1.7, "Position Indicator Channels").

The relative position indication is processed by a Programmable Logic Controller (PLC) which counts sequential electrical pulses sent to the CRD motor stator. There is one relative position transducer (absolute position or relative position is selectable for display on one position indication meter) for each CONTROL ROD drive. Individual rods in a group receive the same signal to move; therefore, the counters for all rods in a group should normally indicate the same position. The Relative Position Indicator System is considered highly precise (one rotation of the roller nut assembly will result in ¾ inch in rod motion). However, if a rod does not move for each demand pulse, the PLC will still count the pulse and incorrectly reflect the position of the stuck (or mechanically constrained) rod.

The Absolute Position Indicator System provides an accurate indication of actual CONTROL ROD position, but at a lower precision than the relative position indicators. This system is based on analog signals from a series of reed switches.

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability are analyzed in the safety analysis (Ref. 3). The criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary damage; and

b. The core must remain subcritical after accident transients, except for a main steam line break (MSLB). The analysis results for a MSLB with a coincident failure of the most reactive rod to insert results in a return to criticality.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 when one CONTROL ROD drops partially or fully into the reactor core. With ICS in manual, this event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod position limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3).

The CONTROL ROD group alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The limits on CONTROL ROD group alignment, safety rod position, and APSR alignment, together with the limits on regulating rod position, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs ensure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted, so that excessive local LHRs will not occur and the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is 6.5% (9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the equipment used to determine this value. For the purpose of complying with this LCO, the position of a misaligned rod is not included in the calculation of the rod group average position.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDM or ejected rod worth, all of which may constitute initial conditions inconsistent with the safety analysis.

BASES (continued)

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant 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 unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and resultant local power peaking would not exceed fuel design limits. In MODES 3, 4, 5 and 6, the OPERABILITY of the 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 MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

ACTIONS

A.1

Alignment of the inoperable or misaligned CONTROL ROD must be accomplished by moving the single CONTROL ROD to the group average position to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and position limits of LCO 3.2.1, "Regulating Rod Position Limits," given in the COLR. THERMAL POWER must also be restricted, as necessary, to the value allowed by the position limits of LCO 3.2.1. The required Completion Time of 1 hour is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR.

A.2.1.1

Compliance with Required Actions of Condition A allows for continued power operation with one CONTROL ROD declared inoperable due to inoperable position indication but trippable, or misaligned from its group average position. These Required Actions comprise the final alternate for Condition A.

If realignment of the CONTROL ROD to the group average is not completed within 1 hour (Required Action A.1 not met), the rod shall be considered inoperable. Since the rod may be inserted farther than the group average position for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement specified in the COLR within 1 hour and once per 12 hours thereafter is adequate to determine that SDM requirements are met.

BASES

ACTIONS (continued)

A.2.1.2

Restoration of the required SDM requires increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate 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 for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

A.2.2

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

A.2.3

Reduction of the nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2.1 to adjust the trip setpoints.

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.18% Δk/k at RTP, 0.36% Δk/k at 80% RTP, or 0.7% Δk/k at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is

BASES

ACTIONS

A.2.4 (continued)

acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 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 for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored. If more than one trippable CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

BASES

ACTIONS

C.1.2 (continued)

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

D.1.1 and D.1.2

When one or more rods are untrippable, the 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 to restore SDM.

In this situation, SDM verification must include the worth of the untrippable rod as well as a rod of maximum worth.

D.2

If the untrippable rod(s) cannot be restored to OPERABLE status, 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 12 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 unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits allows the operator to detect a rod that is beginning to deviate from its expected position. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 could result in radial tilts. Exercising each individual CONTROL ROD 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 an amount in any direction sufficient to demonstrate the absence of mechanical binding will not cause radial or axial power tilts, or oscillations, to occur. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. 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 is determined to be trippable and aligned, 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 CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The rod drop time given in the safety analysis is 1.66 seconds at reactor coolant full flow conditions and $\leq$ 1.40 seconds at no flow conditions to ¾ insertion (Ref. 5). The zone reference switch will activate at ¾ insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. Measuring CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or CONTROL ROD drop time. This Surveillance is performed during a unit outage, due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

This testing is normally performed with all reactor coolant pumps operating to simulate a reactor trip under actual conditions.

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36.
5. UFSAR, Section 15.7.3.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and APSR alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are ONS Design Criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation within design power peaking limits.

Limits on APSR alignment and OPERABILITY have been established, and all APSR positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved.

APSRs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the roller nut assembly, but at different rates (jog and run) depending on the signal output from the Rod Control Drive System.

The APSRs are arranged into a group that is radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip.

APPLICABLE SAFETY ANALYSES

There are no explicit safety analyses associated with mis-aligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR misalignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 assumes the rods are aligned.

During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive

BASES

APPLICABLE SAFETY ANALYSES (continued)

power peaking. The reload safety evaluations define APSR alignment limits that allow APSRs to be positioned anywhere within the operating band and the increase in local LHR is within the design limits. The Required Actions provide a conservative approach to ensure that continued operation remains within the bounds of the safety analysis. No safety analyses take credit for movement of the APSRs.

The APSR alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The limits on CONTROL ROD group alignment, safety rod position, and APSR alignment, together with the limits on regulating rod position, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is 6.5% (9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the equipment used to determine this value.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, and LHRs, which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, when the APSRs are not fully withdrawn because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of APSRs have the potential to affect the safety of the unit. OPERABILITY and alignment of the APSRs are not required when they are fully withdrawn because they do not influence core power peaking. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and excessive local LHRs cannot occur from APSR misalignment.

ACTIONS

A.1

The ACTIONS described below are required if one APSR is declared inoperable due to inoperable position indication or is misaligned. The unit is not allowed to operate with more than one inoperable or misaligned APSR. This would require the reactor to be placed in MODE 3, in accordance with LCO 3.0.3.

BASES

ACTIONS

A.1 (continued)

The reactor may continue in operation with the APSR inoperable or misaligned if the limits on AXIAL POWER IMBALANCE are surveilled within 2 hours to determine if the AXIAL POWER IMBALANCE is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the AXIAL POWER IMBALANCE surveillance to be performed again within 2 hours after each APSR movement. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

B.1

The unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems. In MODE 3, APSR alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 3.1.

2. 10 CFR 50.46.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL

B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

According to ONS Design Criteria (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD alignment and position limits and APSR alignment limits.

The OPERABILITY of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the CONTROL RODS is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM.

Mechanical or electrical failures may cause a CONTROL ROD or APSR to become misaligned from its group. CONTROL ROD or APSR misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design power peaking limits and the core design requirement of a minimum SDM. CONTROL ROD and APSR position indication is needed to assess rod OPERABILITY and alignment.

Limits on CONTROL ROD and APSR alignment and group position have been established, and all CONTROL ROD and APSR positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Two methods of CONTROL ROD and APSR position indication are provided in the Rod Drive Control System. The two means are by absolute and relative position indicator instrumentation. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the control rod drive mechanism (CRDM) motor tube extension.

BASES

BACKGROUND (continued)

Switch contacts close when a permanent magnet mounted on the upper end of each CONTROL ROD and APSR assembly (CRA) leadscrew extension comes near. As the leadscrew and CONTROL ROD or APSR move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and absolute position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators. The relative position indication is processed by a Programmable Logic Controller (PLC) that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

The type R4C absolute position indicator design is used. The type R4C (redundant four channel) absolute position indicator transducer has two parallel sets of voltage divider circuits made up of 36 resistors each, connected in series (channels A and B). One end of 36 reed switches is connected at a junction between each of the resistors of the two parallel circuits. The reed switches making up each circuit are offset, such that the switches for channel A are staggered with the switches for channel B. The type R4C is designed such that either two or three reed switches are closed in the vicinity of the magnet. By its design, the type R4C absolute position indicator provides redundancy, with the two - three sequence of pickup and drop out of reed switches to enable a continuity of position signal when a single reed switch fails to close.

CONTROL ROD and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel. A selector switch permits either relative or absolute position indication to be displayed. Indicator lights are provided on the position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, or enabled, and whether a rod position asymmetry alarm condition is present. Alternate indicators show full insertion, full withdrawal, and under control for each CONTROL ROD and APSR group.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Position Limits" and LCO 3.2.1, "Regulating Rod Position Limits").

BASES

APPLICABLE SAFETY ANALYSES (continued)

CONTROL ROD and APSR positions must be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits" and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the unit is operating within the bounds of the accident analysis assumptions.

The CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

LCO 3.1.7 specifies that one position indicator channel be OPERABLE for each CONTROL ROD and APSR.

This requirement ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits.

APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating THERMAL POWER.

ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore, LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.1.7.1 A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. This CHANNEL CHECK will detect gross failures. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. UFSAR, Section 3.1. 2. UFSAR, Chapter 15. 3. 10 CFR 50.36.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

The SR is modified by a Note indicating that it is only applicable to Unit(s) with the ESPS digital upgrade complete. This SR requires manual actuation of the output channel interposing relays (referred to as Ro relays) to demonstrate OPERABILITY of the relays. The proper functioning of the processor portion of the channel is continuously checked by automatic cyclic self monitoring.

Failure of reactor building purge valves PR-1, 2, 3, 4, 5, 6 to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these 48 inch purge lines. Because of their large size, the 48 inch purge valves are not qualified for automatic closure from their open position under accident conditions. Therefore, the 48 inch purge valves are maintained sealed closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained (Reference 4). Since they are sealed closed in all modes where the Engineered Safeguards system is required operable, testing of these reactor building purge valves is not required per SR 3.3.7.1.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.2

SR 3.3.7.2 is the performance of a CHANNEL FUNCTIONAL TEST. *For Unit(s) with the ESPS digital upgrade complete, the functional test consists of rebooting the digital processors. This verifies that the software has not changed.*

Failure of reactor building purge valves PR-1, 2, 3, 4, 5, 6 to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these 48 inch purge lines. Because of their large size, the 48 inch purge valves are not qualified for automatic closure from their open position under accident conditions. Therefore, the 48 inch purge valves are maintained sealed closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained (Reference 4). Since they are sealed closed in all modes where the Engineered Safeguards system is required operable, testing of these reactor building purge valves is not required per SR 3.3.7.2. This applies to units with the ESPS digital upgrade complete and not complete.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.7.2 (continued)

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The digital ESPS software performs a continuous online automated cross channel check, separately for each channel, and continuous online signal error detection and validation. The protection system also performs continual online hardware monitoring. The CHANNEL FUNCTIONAL TEST essentially validates the self monitoring function and checks for a small set of failure modes that are undetectable by the self monitoring function.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.
4. NUREG 0737, Section II.E.4.2.6.

B 3.7 PLANT SYSTEMS

B 3.7.16 Control Room Area Cooling Systems (CRACS)

BASES

BACKGROUND

The CRACS provides temperature control for the control areas.

The control area is defined as the control room, cable room, and equipment room for each unit. Units 1 and 2 have a shared control room, and Unit 3 has an independent control room. The cable and equipment rooms are independent for each unit. The control rooms, cable rooms, and equipment rooms for each unit contain vital electrical equipment, such as 125 VDC Vital I&C Power and 120 VAC Vital I&C Power, which is essential for achieving safe shutdown on the units. A control area portion is defined as a cable room, equipment room, or control room, for which a set of redundant CRVS cooling trains is required. The control area portions are listed in the table below. Through the use of alternative air flow paths, air handling units AHU-34 and AHU-35 provide redundant cooling to both Units 1 and 2 cable rooms.

The AHUs which cool the control areas are part of the CRVS for each unit. The Chilled Water System (WC) serves as the heat sink for the CRVS on all three units. The WC System consists of two redundant cooling trains which serve all three units.

UFSAR Section 9.4.1 (Ref. 1) requires that redundant air conditioning and ventilation equipment be available to assure that no single failure of an active component within the CRVS and WC System will prevent proper control area environmental control. During a LOOP event, power will be temporarily lost to the equipment within these systems. Upon restoration of power the equipment will be required to restart. This restart makes the equipment susceptible to a single active failure. Without redundant cooling capability, acceptable temperatures within the control area could be exceeded. This could result in the potential failure of vital electrical equipment which is needed for safe shutdown of the units.

BASES

BACKGROUND (continued)

The following table identifies each portion of the CRVS where redundancy is required:

Table B 3.7.16-1
CRVS Redundant Equipment

Control Area Portion	Associated CRVS Cooling Trains
Unit 1&2 Control Room	AHU-11 and AHU-12
Unit 1 Cable Room	AHU-34 and AHU-35
Unit 1 Equipment Room	AHU-22 and AHU-34
Unit 2 Cable Room	AHU-34 and AHU-35
Unit 2 Equipment Room	AHU-23 and AHU-35
Unit 3 Control Room	AHUs 3-13 and 3-14
Unit 3 Cable Room	AHUs 3-11 and 3-12
Unit 3 Equipment Room	AHUs 3-15 and 3-16

A single train will provide the required temperature control. The CRACS operation to maintain control room temperature is discussed in the UFSAR, Section 9.4.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CRACS is to maintain control area temperature to ensure cooling of vital equipment.

The CRACS components are arranged in redundant trains. A single active failure of a CRACS component does not impair the ability of the system to perform as designed. The CRACS is designed to remove sensible and latent heat loads from the control area, including consideration of equipment heat loads to ensure equipment OPERABILITY.

The CRACS satisfies Criterion 3 of the NRC Policy Statement.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the heat removal capability of the system is sufficient to maintain the temperature in the control room and cable room at or below 80°F and maintain the temperature in the electrical equipment room at or below 85°F. The temperature is determined by reading gauges in each area or computer points which are considered representative of the average area temperature. These temperature limits are based on operating history and are intended to provide an indication of degradation of the cooling systems. The limits are conservative with respect to equipment operability temperature limits. The values for the SR are values at which the system is removing sufficient heat to meet design requirements (i.e., OPERABLE) and sufficiently above the values associated with normal operation during hot weather. The temperature in the equipment room is typically slightly higher than the temperature in the control room or cable room. Because of that, a higher value is specified for this area. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 9.4.1.