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LIC-12-0006
February 10, 2012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Reference: Docket No. 50-285

SUBJECT: Fort Calhoun Station (FCS) License Amendment Request (LAR) 12-01, Proposed Change to Establish the Reactor Protective System (RPS) Actuation Circuits Limiting Condition for Operation (LCO)

Pursuant to 10 CFR 50.90, the Omaha Public Power District (OPPD) hereby requests an amendment to the Renewed Facility Operating License No. DPR-40 for Fort Calhoun Station (FCS), Unit No. 1. The proposed amendment would establish the limiting condition for operation (LCO) requirements for the reactor protective system (RPS) actuation circuits in Technical Specification (TS) 2.15.

These TS revisions will result in the TS LCOs and surveillance requirements (SRs) being more aligned with NUREG-1432, *Standard Technical Specifications, Combustion Engineering Plants*, Revision 3, for RPS requirements. Specifically, this proposed change: renumbers LCO 2.15(1) through 2.15(4) to 2.15.1(1) through 2.15.1(4), renumbers LCO 2.15(5) to LCO 2.15.3 with an associated Table 2-6, and implements a new LCO 2.15.2 for the reactor protective system logic and trip initiation channels.

The Table of Contents is also revised to reflect the renumbering and addition of the LCO for the reactor protective system logic and trip initiation channels and the new Table 2-6. Currently, the TS contain surveillance requirements in TS 3.1, Table 3-1, Item 12, for a quarterly functional test of the RPS logic units, and TS 3.1 Table 3-1, Item 10 for a prior to critical functional test of manual trips, but contains no LCO for the RPS logic units. These TS revisions will result in the TS LCOs and SRs for the RPS logic units and manual trips being similar to the Palisades plant which has a similar design for the reactor trip initiation channels.

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The proposed TS changes conform to NRC regulation 10 CFR 50.36 for the contents of the Technical Specifications.

OPPD concludes that the proposed LAR presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The enclosure contains OPPD's evaluation of the proposed changes, including the supporting technical evaluation, and the significant hazards consideration determination. Attachment 1 provides the existing TS and TS Bases pages marked-up to show the proposed changes to TS 2.15. Attachment 2 provides the associated retyped (clean) TS and TS Bases pages.

OPPD requests approval of the proposed amendment by March 1, 2013. Once approved, the amendment shall be implemented within 180 days.

There are no regulatory commitments associated with this proposed change.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Nebraska official.

If you should have any questions regarding this submittal or require additional information, please contact Mr. Bill R. Hansher at 402-533-6894.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 10, 2012.

David J. Bannister
Vice President and CNO

DJB/BRH/brh

Enclosure: OPPD's Evaluation of the Proposed Changes

c: E. E. Collins, Jr., NRC Regional Administrator, Region IV
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OPPD's Evaluation of the Proposed Change(s)

Fort Calhoun Station (FCS) License Amendment Request (LAR) 12-01, Proposed Change to Establish the Reactor Protective System (RPS) Actuation Circuits Limiting Condition for Operation (LCO)

ATTACHMENTS:

1. Technical Specification and Information Only Bases Pages Markups
2. Retyped ("Clean") Technical Specifications and Information Only Bases Pages

1.0 SUMMARY DESCRIPTION

This license amendment request (LAR) proposes a change to Renewed Facility Operating License No. DPR-40 for Fort Calhoun Station (FCS), Unit No. 1. The Omaha Public Power District (OPPD) proposes to revise the Technical Specification (TS) limiting condition for operation (LCO) 2.15, *Instrumentation and Control Systems.* Specifically, this proposed change: renumbers LCO 2.15(1) through 2.15(4) to 2.15.1(1) through 2.15.1(4), renumbers LCO 2.15(5) to LCO 2.15.3 with an associated Table 2-6, and implements a new LCO 2.15.2 for the reactor protective system logic and trip initiation channels.

The Table of Contents is also revised to reflect the renumbering and addition of LCO 2.15.2 for the reactor protective system logic and trip initiation channels and the relocation of TS 2.15(5) to a new LCO 2.15.3 and new Table 2-6. Currently, the TS contain surveillance requirements in TS 3.1, Table 3-1, Item 12, for a quarterly functional test of the RPS logic units, and TS 3.1 Table 3-1, Item 10 for a prior to critical functional test of manual trips, but contains no LCO for the RPS logic units. These TS revisions will result in the TS LCOs and SRs for the RPS logic units and manual trips being similar to the Palisades plant which has a similar design for the reactor trip initiation channels.

2.0 DETAILED DESCRIPTION

The proposed TS changes for LAR 12-01 are as follows:

- TS LCO 2.15, *Instrumentation and Control Systems*
  - Renumbered to LCO 2.15.1 for paragraphs (1) through (4) with appropriate renumbering of footnotes in Tables 2-2 through 2-5 that reference these paragraphs.
  - Conversion of LCO 2.15(5) with its list of components into a new LCO 2.15.3 with the list of components being included into a new Table 2-6.
- New TS LCO 2.15.2 for the reactor protective system logic and trip initiation channels.
- Deletion of reactor protective system manual trip functional unit from TS Table 2-2 and inclusion into the new LCO 2.15.2.

These proposed changes to TS 2.15 will result in the FCS LCO 2.15 being more aligned with NUREG 1432, *Standard Technical Specifications, Combustion Engineering Plants* for the RPS.

3.0 TECHNICAL EVALUATION

System Description

The Reactor Protective System (RPS) is shown in Updated Safety Analysis Report (USAR) Figures 7.2-1 and 7.2-2 (References 6.1 and 6.2). It consists of four channels of instrumentation. Each channel monitors 12 safety parameters, each parameter input is derived from an isolated instrument channel. Each parameter operates a two out of four coincidence logic matrix to maintain OR remove power from the Control Element Drive Mechanism (CEDM) clutches. Individual channel trips occur when the measurement reaches a preselected value. A typical measurement channel functional diagram is shown in USAR Figure 7.2-3. The channel trips are combined in six two-out-of-four matrices. Each individual measurement channel has inputs to three of the six logic matrices. The logic matrix trip relays are de-energized when two channels of the same measurement channel trip. Each two-out-of-four logic matrices provides trip signals to the interposing relays which in turn cause a direct trip of the contactors in the a-c supply to the CEDM clutch power supplies. Any one of the six logic matrices will de-energize the 4 clutch power supplies. The logic matrices are arranged in a one-out-of-six logic configuration. The clutch power supply DC outputs are ungrounded.

Reactor trip is accomplished by de-energizing the magnetic clutch holding coils and releasing the control element assemblies (CEAs) to drop into the core.

The Reactor Protective System was designed under the following bases to assure adequate protection for the reactor core (Reference 6.3):

a. Instrumentation conformed to the provisions of the proposed IEEE Standard for Nuclear Power Plant Protective Systems (IEEE 279, August 1968).
b. No single component failure can prevent safety action.
c. Four independent measurement channels, each complete with sensors, sensor power supply units, amplifiers, and bistable modules were provided for each parameter.
d. The channels are provided with a high degree of independence by separate connection of the sensors to the process systems. Separate raceways are used to segregate cable systems.
e. The four measurement channels provide trip signals to four independent trip paths.
f. A trip signal from any two trip units monitoring the same parameter causes a reactor trip.
g. When one of the four channels is taken out of service for maintenance, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the out-of-service channel.
h. The protective system AC power is supplied from four separate instrument buses.
i. Open circuiting, or loss of power supply for the channel logic, initiates an alarm and a channel trip.
j. All measurement channels and trip logic matrices assume the de-energized state to provide a tripping function.

k. The Reactor Protective System can be tested with the reactor in operation or shutdown.
l. A manual trip, independent of the automatic trip system, is provided.
m. Trip signals are preceded by alarms to alert the operator of undesirable operating conditions in cases where the operator could avert a reactor trip by taking timely corrective actions.
n. The Reactor Protective System components are independent of the Power Range Monitor control channels.

Coincidence Logic Matrices

The RPS has four separate channels having twelve trip units per channel. Each of the twelve trip units serves to monitor a different plant parameter. There are four trip units for each plant parameter monitored, one per channel.

There are six logic trip matrices (AB, BC, BD, AC, CD, and AD). For matrix AB, the normally open contact from the channel A trip unit 1 relay 1 (A1-1) is connected in parallel with the channel B trip unit 1 relay 1 (B1-1).

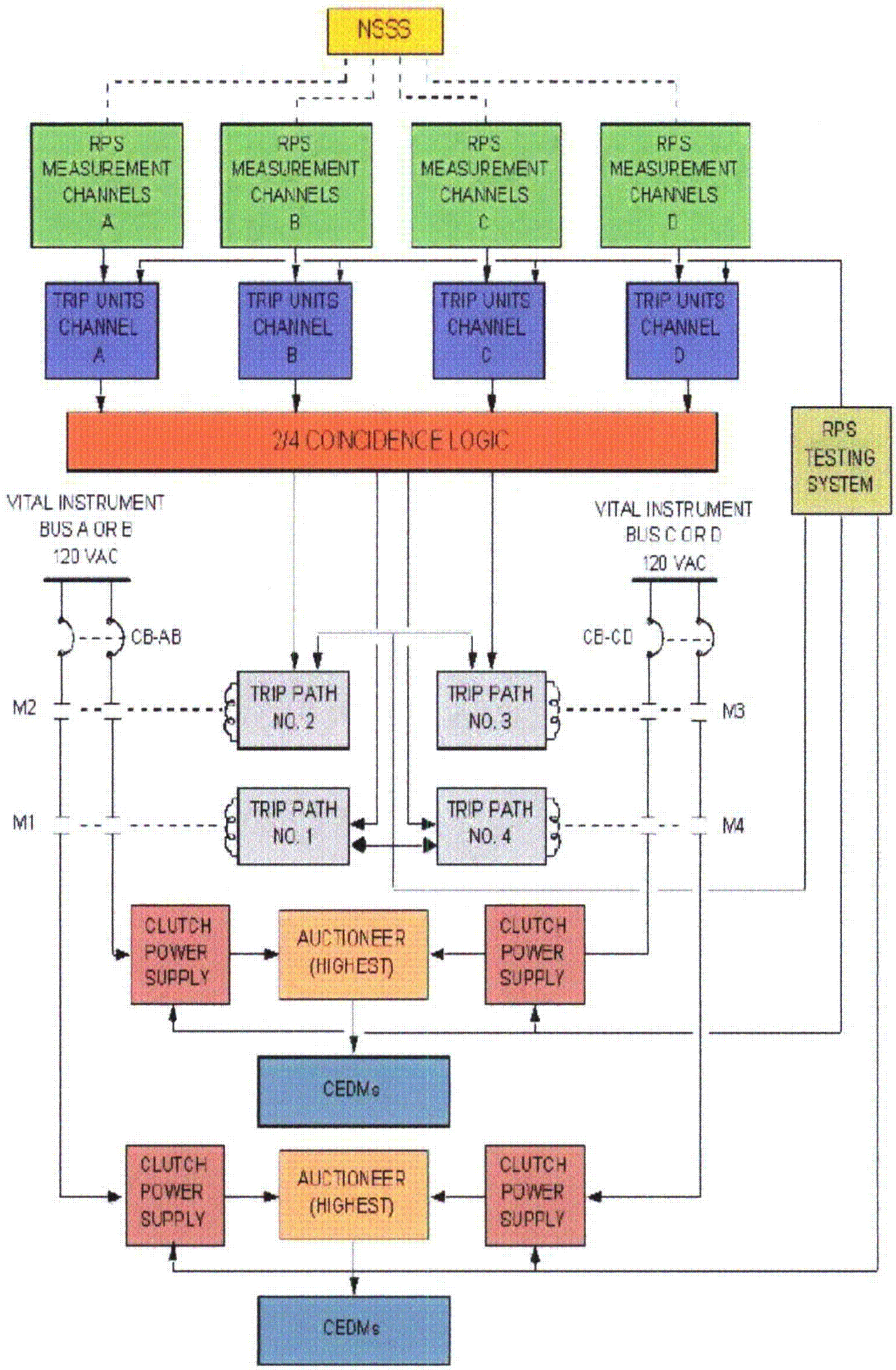
Each trip unit contains three sealed trip relays which have a single-pole, double-throw (SPDT) contact. The normally open contact from channel A trip unit 1 relay 1 (A1-1) is connected in parallel with the normally open contact from channel B trip unit 1 relay 1 (B1-1). This is similarly done for the twelve contact combinations A1-1 through A12-1 and B1-1 through B12-1 and these twelve parallel combinations are connected in series to form a logic ladder. The trip unit relays are energized in a reset condition, thus the normally open relay contacts are closed. The AB logic ladder serves to control four matrix relays which are energized in the reset condition. The three trip unit trip relays from the four channels are used to make six logic matrices in the same fashion as the AB matrix.

A normally open contact from each of the four matrix relays are connected in series with a normally open contact from the corresponding relays of the other five matrices to form four trip paths. A contact from one of the manual trip switches and the coil of an interposing relay are in series with the six matrix relay contacts for each of the four trip paths. Under normal operating conditions the four interposing relays are energized. A normally open contact from each of the interposing relays serves to control an M-contactor.

The four M-contactors are combined into pairs with the contacts of each pair connected in series. The series contacts of the two pairs serve to supply AC power to the CEDM clutch power supplies. The CEDM clutches are separated into two groups. The clutches in each group are supplied by parallel pairs of low voltage, d-c power supplies which are fed by an ungrounded supply via contacts from the two pairs of M-contactors. The parallel CEDM clutch power supplies assure that the inadvertent loss of one supply source will not de-energize the clutches.

A block diagram of the RPS is provided below:

REACTOR PROTECTIVE BLOCK DIAGRAM



Currently, the TS contain surveillance requirements in TS 3.1, Table 3-1, Item 12, for a quarterly functional test of the RPS logic units, and TS 3.1 Table 3-1, Item 10 for a prior to critical functional test of manual trips, but contains no LCO for the RPS logic units. Since there is no LCO, if an RPS logic unit or trip initiation channel becomes inoperable, TS 2.0.1 would apply. TS 2.0.1 specifies corrective measures to be employed for system conditions not covered by LCOs. The corrective measures include placing the unit in Hot Shutdown within 6 hours, in at least subcritical and less than 300 degrees F within an additional 6 hours and placing the unit in Cold Shutdown within the following 30 hours. In certain situations these actions are non-conservative compared to NUREG 1432, (Reference 6.4). As an example, NUREG 1432 requires the reactor trip breakers be opened immediately for certain conditions.

Applicability

The proposed TS would revise the applicability for when the RPS logic and trip initiation channels are required to be operable to include whenever CEAs are capable of being withdrawn and the reactor coolant system is not at refueling boron concentration. The definition section of the TS includes definitions for both CEAs and non-trippable CEAs. As the RPS trip initiation channels do not trip non-trippable CEAs, the LCO only applies to CEAs that are defined as all full length shutdown and regulating control rods. By TS definition, Mode 4 is RCS less than 210 degrees F with a boron concentration of greater than or equal to shutdown boron concentration but less than refueling boron concentration, Mode 5 is RCS less than 210 degrees F with a boron concentration of greater than or equal to refueling boron concentration.

RPS trip functions are not required while in modes of operation when the RCS boron concentration is at refueling boron concentration, or when no more than one trippable control rod is capable of being withdrawn, because the RPS function is already fulfilled. Refueling boron concentration provides sufficient negative reactivity to assure the reactor remains subcritical regardless of control rod position, and the safety analyses assume that the highest worth withdrawn full-length control rod will fail to insert on a trip. Therefore, under these conditions, the safety analyses assumptions will be met without the RPS trip function.

Specification

The proposed TS would apply to the six channels of RPS logic matrices and four channels of RPS trip initiation channels which do not currently have a separate LCO, and the two channels of manual trip initiation which are currently contained in TS 2.15, Table 2-2, Item 1.

Required Actions

With one RPS Logic Matrix channel inoperable, it is proposed to restore the inoperable channel to operable status within 48 hours. The completion time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable.

With one RPS Manual Trip channel inoperable, the required action is to restore the inoperable channel to operable status prior to entering Mode 2 from Mode 3. This action is consistent with the current TS requirement contained in TS Table 2-2, Item 1. There are two installed Manual Trip channels. TS 2.15, Table 2-2, Item 1 requires that there be at least one operable. No safety analyses assume operation of the Manual Trip. Because of this, the required action is to restore the inoperable channel to operable status prior to entering Mode 2 from Mode 3 during the next plant startup.

With one RPS Trip Initiation logic channel inoperable, it is proposed to de-energize the affected clutch power supply within one hour. This removes the need for the affected channel by performing its associated safety function. With the clutch power supplies associated with one initiation logic channel de-energized, the remaining clutch power supplies prevent control rod clutches from de-energizing. The remaining clutch power supplies are in a one-out-of-two logic with respect to the remaining initiation logic channels in the clutch power supply path. This meets redundancy requirements, but testing on the operable channels cannot be performed without causing a reactor trip.

With two inoperable RPS Trip Initiation logic channels affecting the same trip leg, it is proposed to de-energize the affected clutch power supplies immediately. With both channels inoperable, the RPS function is lost if the affected clutch power supplies are not de-energized. Therefore, immediate action is required to de-energize the affected clutch power supplies. The immediate completion time is appropriate since there could be a loss of safety function if the associated clutch power supplies are not de-energized.

With the required actions not met, or with two RPS Manual Trip channels inoperable or with two or more RPS logic matrices inoperable, or with two or more RPS Trip Initiation logic channels inoperable that do not affect the same trip leg, it is proposed that within 6 hours; the station is placed in Hot Shutdown, it is verified that no more than one CEA can be withdrawn, or the RCS is verified to be at refueling boron concentration. In this condition the reactor must be placed in a Mode in which the LCO does not apply. The completion time of 6 hours to be in Mode 3 is reasonable, based on operating experience, to reach the required Mode from full power conditions in an orderly manner and without challenging plant systems.

The requirements of TS 2.15(5) are being incorporated into a new LCO 2.15.3 with the list of components being included into a new Table 2-6. No changes are proposed for the requirements other than formatting to be more consistent with the remainder of TS 2.15 by listing the components in a Table.

These TS revisions will result in the TS LCOs and SRs for the RPS logic units and manual trips being similar to the Palisades plant which has a similar design for the reactor trip initiation channels.

10 CFR Part 50.36 Criteria:

10 CFR 50.36(c)(2)(ii) states that "A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) Criterion 1 - Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2 - A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) Criterion 3 - A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4 - A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

NRC-approved NUREG 1432, *Standard Technical Specifications, Combustion Engineering Plants*, Revision 3 (Reference 6.4), identifies an improved TS that was developed based on the screening criteria in the Commission's Final Policy Statement (Reference 6.5) and subsequently codified in 10 CFR 50.36. The RPS actuation circuits satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (described above) and their operability will be required by the proposed changes to FCS TS 2.15.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Regulations

Code of Federal Regulations Part 50:

10 CFR 50.36, *Technical Specifications*: 10 CFR 50.36(c)(2) states, "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." The revised actions will establish limiting conditions for operation for the RPS logic matrices and RPS trip initiation channels which do not currently have a separate LCO and therefore the proposed change will meet the requirements of this regulation.

10 CFR 50.36(c)(3) criteria states that "surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

The revised actions will maintain the surveillance requirements on the RPS logic units and manual trip functions and therefore the proposed change will continue to meet the requirements of this regulation.

Fort Calhoun Station (FCS), Unit No. 1 was licensed for construction prior to May 21, 1971, and is committed to the draft General Design Criteria (GDC) published for comment in the Federal Register on July 11, 1967 (32 FR 10213) in lieu of 10 CFR 50, Appendix A. Appendix G of the FCS Updated Safety Analysis Report (USAR) shows that draft GDC 12, 14, 19, and 25 are most applicable to the proposed amendment.

<u>CRITERION 12 - *INSTRUMENTATION AND CONTROL SYSTEMS*</u>

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

This criterion is met. Instrumentation is provided for continuous measurement of all significant process variables. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation. The instrumentation conforms to applicable Institute of Electrical and Electronics Engineers (IEEE) standards.

The principal process variables monitored include neutron level (reactor power); reactor coolant temperature, flow, and pressure; pressurizer liquid level; and steam generator level. In addition, instrumentation is provided for continuous automatic monitoring of radiation level. The instrumentation and control systems are described in detail in USAR Section 7.

The proposed license amendment request provides for addition of the RPS logic operability LCO. No physical changes are being made to the plant. This criterion will continue to be met.

<u>CRITERION 14 – CORE PROTECTION SYSTEMS</u>

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

This criterion is met. The reactor is protected by the reactor protection system from reaching a condition at which fuel damage might occur. The protection *system is designed to monitor the reactor operating conditions and initiate a fast shutdown if any of measured variables exceed the operating limits.*

The signals which will provide automatic reactor trip are identified in Table 7.2-1 of the Fort Calhoun Station, Unit No. 1 USAR. The parameters and conditions which will initiate a trip are the following:

a) High Neutron Level (reactor power) (ΔT power is a backup)
b) High Startup Rate (low power level only)
c) High Pressurizer Pressure
d) Thermal Margin/Low Pressure
e) Loss of Load
f) Low Steam Generator Pressure
g) Low Reactor Coolant Flow
h) Low Steam Generator Liquid Level
i) Containment Building High Pressure
j) Steam Generator Differential Pressure

The proposed license amendment request provides for addition of the RPS logic operability LCO. Applicability of the LCO will ensure that the RPS is capable of being tripped whenever a CEA is capable of being withdrawn and the RCS is not at refueling boron concentration. No physical changes are being made to the plant. This criterion will continue to be met.

CRITERION 19 – PROTECTION SYSTEMS RELIABILITY

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

This criterion is met. Design of protection systems includes specification of high quality components, ample design capacity, component redundancy, and in-service testability. The following principal design criteria have been applied:

a) No single component failure shall prevent the protection systems from fulfilling their protective function when action is required.

b) No single component failure shall initiate unnecessary protection system action provided implementation does not conflict with the criterion above.

Testing facilities are built into the protection systems to provide for:

a) Preoperational testing to give assurance that the protection systems can fulfill their required functions.

b) In-service checking of protective channels from the process sensor to the channel trip unit (bistable).

c) *In-service testing of the channel trip units (bistables) and associated coincidence logic and the outputs of that logic through to the final actuator.*

The proposed license amendment request provides for addition of the RPS logic operability LCO. Surveillance requirements to perform quarterly functional tests of the RPS logic units remain unchanged. No physical changes are being made to the plant. This criterion will continue to be met.

CRITERION 25 – DEMONSTRATION OF FUNCTION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEM

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

This criterion is met. Protection systems, from the sensors up to the final protection element, will be capable of being checked during reactor operation, as follows:

a) *Measurement channels used in protection systems will be checked by observing outputs of similar channels and cross checking with related measurements which are presented on indicators and recorders on the control board.*

b) *Trip units and logic will be tested by inserting a signal into the measurement channel ahead of the readout and, upon application of a trip level input, observing that a signal is passed through the trip units and the logic to the logic output relays.*

c) *The logic output relays will be tested individually for initiation of trip action.*

The proposed license amendment request provides for addition of the RPS logic operability LCO. No physical changes are being made to the plant. Currently, the TS contain surveillance requirements in TS 3.1, Table 3-1, Item 12, for a quarterly functional test of the RPS logic units, and TS 3.1 Table 3-1, Item 10 for a prior to critical functional test of manual trips. No changes to these surveillance requirements are being made. This criterion will continue to be met.

4.1.2 Design Basis

The design basis requirement for the RPS was discussed in the Technical Evaluation Section 3.0 as it relates to the RPS logic and trip functions and their associated USAR sections.

4.1.3 Approved Methodologies

There are no new specific approved methodologies associated with this proposed TS change.

4.1.4 Analysis

No new analyses were needed in support of this proposed LAR.

4.2 Precedent

As noted in Section 3.0 above, precedent for adding the TS LCOs and SRs for RPS logic and trip initiation circuits is similar to NUREG 1432 for the reactor trip breakers. The design of the FCS RPS is similar to that of the Palisades plant in that there are M-contactors instead of reactor trip breakers. This submittal is consistent with these documents; however, no plant-specific precedence is cited in this LAR.

4.3 Significant Hazards Consideration

The proposed change would modify Technical Specification (TS) 2.15 to include provisions for the reactor protective system logic and trip initiation circuits. These proposed changes are aligned with NUREG 1432, *Standard Technical Specifications, Combustion Engineering Plants*, Revision 3. In addition, administrative changes are being made to TS 2.15(5), in that the TS is being reformatted to list components in a table consistent with the format of the remainder of TS 2.15.

The Omaha Public Power District (OPPD) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The reactor protective system logic and trip initiation channels meets Criterion 3 of 10 CFR 50.36 for inclusion into Technical Specification (TS) as a component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient. The TSs currently do not have limiting conditions for operation (LCO) specific for this circuitry, but does contain surveillance requirements. The addition of LCOs provides additional restrictions on the operation of the plant and provides required actions and time limits if these components are incapable of performing their function. As such, the proposed change does not increase the probability of an accident. The proposed changes do not alter the physical design of the RPS, or any other plant structure, system or component (SSC) at Fort Calhoun Station (FCS).

The proposed changes conform to the Nuclear Regulatory Commission's (NRC's) regulatory guidance regarding the content of plant TS as identified in 10 CFR 50.36 and NRC publication NUREG 1432.

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

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed TS changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Hence, the proposed changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure or system in the performance of their safety function.

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

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

The TS operability requirements for the RPS logic and trip initiation channels ensure there is adequate components operable to assure safe reactor operation and are necessary to ensure safety systems accomplish their safety function for design basis accident events. The proposed TS would revise the applicability for when the RPS logic and trip initiation channels are required to be operable to include whenever control element assemblies (CEAs) are capable of being withdrawn and the reactor coolant system (RCS) is not at refueling boron concentration. When the RCS boron concentration is at refueling boron concentration, or when no more than one trippable control rod is capable of being withdrawn, the RPS function is already fulfilled. These proposed TS changes for the RPS are aligned with the applicability and operability requirements provided in NUREG 1432.

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

Based on the above, OPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 USAR, Figure 7.2-1, Reactor Protective System

6.2 USAR, Figure 7.2-2, Reactor Protective System Functional Diagram

6.3 USAR Section 7.2, Instrumentation and Control, Reactor Protective Systems, Revision 14, dated August 19, 2010

6.4 NUREG 1432, Revision 3, Standard Technical Specifications, Combustion Engineering Plants, dated June 2004

6.5 NRC "Final Policy Statement on Technical Specifications Improvement for Nuclear Power Reactors" (58 FR 39132, dated July 22, 1993)

Technical Specifications and Information Only Bases Pages Markups

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TECHNICAL SPECIFICATIONS - TABLES

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2.0 LIMITING CONDITIONS FOR OPERATION

2.15.1 Instrumentation and Control Systems

Applicability

Applies to plant instrumentation systems.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specifications

The operability, permissible bypass, and Test Maintenance and Inoperable bypass specifications of the plant instrument and control systems shall be in accordance with Tables 2-2 through 2-5.

(1) In the event the number of channels of a particular system in service falls one below the total number of installed channels, the inoperable channel shall be placed in either the bypassed or tripped condition within one hour if the channel is equipped with a bypass switch, and eight hours if jumpers or blocks must be installed in the control circuitry. The inoperable channel may be bypassed for up to 48 hours from time of discovering loss of operability; however, if the inoperability is determined to be the result of malfunctioning RTDs or nuclear detectors supplying signals to the high power level, thermal margin/low pressurizer pressure, and axial power distribution channels, these channels may be bypassed for up to 7 days from time of discovering loss of operability. If the inoperable channel is not restored to OPERABLE status after the allowable time for bypass, it shall be placed in the tripped position or, in the case of malfunctioning RTDs or linear power nuclear detectors, the reactor shall be placed in hot shutdown within 12 hours. If active maintenance and/or surveillance testing is being performed to return a channel to active service or to establish operability, the channel may be bypassed during the period of active maintenance and/or surveillance testing. This specification applies to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(2) In the event the number of channels of a particular system in service falls to the limits given in the column entitled "Minimum Operable Channels," one of the inoperable channels must be placed in the tripped position or low level actuation permissive position for the auxiliary feedwater system within one hour, if the channel is equipped with a bypass switch, and within eight hours if jumpers or blocks are required; however, if minimum operable channel conditions for SIRW tank low signal are reached, both inoperable channels must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If at least one inoperable channel has not been restored to OPERABLE status after 48 hours from time of discovering loss of operability, the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the containment ventilation isolation valves are closed.

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2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15.1 Instrumentation and Control Systems (Continued)

If after 24 hours from time of initiating a hot shutdown procedure at least one inoperable engineered safety features or isolation functions channel has not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. This specification applied to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(3) In the event the number of channels on a particular engineered safety features (ESF) or isolation logic subsystem in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," sufficient channels shall be restored to OPERABLE status within 48 hours so as to meet the minimum limits or the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure sufficient channels have not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours.

(4) In the event the number of channels of those particular systems in service not described in (3) above falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of OPERABLE high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above 10^{-4}% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

If during power operation, the rod block function of the secondary CEA position indication system and rod block circuit are inoperable for more than 24 hours, or the plant computer PDIL alarm, CEA group deviation alarm and the CEA sequencing function are inoperable for more than 48 hours, the CEAs shall be withdrawn and maintained at fully withdrawn and the control rod drive system mode switch shall be maintained in the off position except when manual motion of CEA Group 4 is required to control axial power distribution.

~~(5) In the event that the number of operable channels of the listed Alternate Shutdown Panels or the Auxiliary Feedwater Panel instrumentation or control circuits falls below the required number of channels, either restore the required number of channels to OPERABLE status within seven (7) days, or be in hot shutdown (Mode 3) within the next twelve hours. This specification is applicable in Modes 1 and 2.~~

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems (Continued)
2.15.2 Reactor Protective System (RPS) Logic and Trip Initiation

~~Function/Instrument or Control Parameter~~	~~Location~~	~~Required Number of Channels~~
~~1. Reactivity Control~~		
~~a. Source Range Power~~	~~AI-212~~	~~1~~
~~b. Reactor Wide Range Logarithmic Power~~	~~AI-212~~	~~1~~
~~2. Reactor Coolant System Pressure Control~~		
~~a. Pressurizer Wide Range Pressure (0-2500 psia)~~	~~AI-179~~	~~1~~
~~3. Decay Heat Removal via Steam Generators~~		
~~a. Reactor Coolant Hot Leg Temperature~~	~~AI-185~~	~~1 (Note 1)~~
~~b. Reactor Coolant Cold Leg Temperature~~	~~AI-185~~	~~1 (Note 1)~~
~~c. Steam Generator Pressure~~	~~AI-179~~	~~1 per Steam Generator~~
~~d. Steam Generator Narrow Range Level~~	~~AI-179~~	~~1 per Steam Generator~~
~~e. Steam Generator Wide Range Level~~	~~AI-179~~	~~1 per Steam Generator~~
~~4. Reactor Coolant System Inventory Controls~~		
~~a. Pressurizer Level~~	~~AI-185~~	~~1~~
~~b. Volume Control Tank Level~~	~~AI-185~~	~~1~~
~~c. Charging Pump CH-1B and its associated controls~~	~~AI-185~~	~~1~~
~~d. Charging Isolation Valve Control~~	~~AI-185~~	~~1~~
~~5. Transfer Functions~~		
~~a. All Transfer Switches/Lockout Relays~~	~~AI-185~~	~~1~~
~~b. All Transfer Switches/Lockout Relays~~	~~AI-179~~	~~1~~

~~Note 1~~~~: One reactor coolant hot leg temperature indication and one reactor coolant cold leg temperature indication channel must both be operable on the same steam generator (i.e., RC-2A or RC-2B)~~.

Applicability

Applies to the operational status of RPS Logic and Trip Initiation channels in MODES 1 and 2; and, When reactor coolant temperature (T_{cold}) is greater than 210°F or MODE 4 with more than one CEA rod capable of being withdrawn and RCS boron concentration less than REFUELING BORON CONCENTRATION.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specification

Six channels of RPS Logic matrices, four channels of RPS Trip Initiation Logic and two channels of RPS Manual Trip shall be OPERABLE.

Required Actions

(1) With one RPS Logic Matrix channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours.

(2) With one RPS Trip Initiation Logic channel inoperable, de-energize the affected clutch power supply within one hour.

(3) With one RPS Manual Trip channel inoperable, restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3.

(4) With two RPS Trip Initiation Logic channels affecting the same trip leg inoperable, de-energize the affected clutch power supplies immediately.

(5) With the required actions of (1), (2), or (4) not met, or
with two RPS Manual Trip channels inoperable, or
with two or more RPS Logic Matrices inoperable, or
with two or more RPS Trip Initiation Logic channels inoperable for reasons other than (4):

a. be in HOT SHUTDOWN and verify no more than one CEA is capable of being withdrawn within 6 hours,

OR

b. verify reactor coolant boron concentration is at REFUELING BORON CONCENTRATION within 6 hours.

2.0 LIMITING CONDITIONS FOR OPERATION

2.15 Instrumentation and Control Systems (Continued)

2.15.3 Alternate Shutdown and Auxiliary Feedwater Panel

~~Function/Instrument or Control Parameter~~	~~Location~~	~~Required Number of Channels~~
~~6. Auxiliary Feedwater Controls~~		
~~a. Steam Generator RC-2A and 2B Auxiliary Feedwater Isolation Inboard and Outboard Valves Control~~	~~AI-179~~	~~1~~
~~b. Steam-Driven Pump FW-10 Recirculation Valve Control~~	~~AI-179~~	~~1~~
~~c. Steam-Driven Pump FW-10 Steam Isolation Valve Control~~	~~AI-179~~	~~1~~
~~d. Steam from Steam Generator RC-2A and RC-2B to FW-10 Steam Isolation Valve Control~~	~~AI-179~~	~~1~~

~~Basis~~

~~During plant operation, the complete instrumentation systems will normally be in service. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protective system (RPS) and engineered safety features (ESF) system when one or more of the channels are out of service. Reactor safety is provided by RPS, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by contined operation with certain instrumentation channels out of service since provisions were made for this in the plant design.~~

~~The RPS and most engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in the ESF logic system.~~

~~When one of the four channels is taken out of service for maintenance, RPS logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),(1) which results in a one-out-of-three channel logic. If in the 2-out-of-4 logic system of the RPS one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1-out-of-2. At rated power, the minimum OPERABLE high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are OPERABLE, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.~~

Applicability

Applies to the operational status of Alternate Shutdown and Auxiliary Feedwater Panel Functions in MODES 1 and 2.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specification

The Alternate Shutdown and Auxiliary Feedwater Panel Functions/Instrumentation or Control Parameters in Table 2-6 shall be OPERABLE.

Required Actions

(1) With the number of OPERABLE channels or control circuits less than the required number of channels, restore the required number of channels to OPERABLE within seven (7) days.

(2) With the required actions of (1) not met, be in HOT SHUTDOWN within 12 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems

Basis

During plant operation, the complete instrumentation systems will normally be in service. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protective system (RPS) and engineered safety features (ESF) system when one or more of the channels are out of service. Reactor safety is provided by RPS, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continued operation with certain instrumentation channels out of service since provisions were made for this in the plant design.

The RPS and most engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in the ESF logic system.

When one of the four channels is taken out of service for maintenance, RPS logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),[(1)] which results in a one-out-of-three channel logic. If in the 2-out-of-4 logic system of the RPS one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1-out-of-2. At rated power, the minimum OPERABLE high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are OPERABLE, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

An RPS Logic matrix channel consists of two matrix power supplies, four matrix relays and their associated contacts as well as all interconnecting wiring. An RPS Trip Initiation Logic channel consists of an M contactor and associated contacts, an interposing relay and all interconnecting wiring. Two RPS Trip Initiation Logic channels associated with the same pair of CEDM clutch power supplies are considered to affect the same trip leg.

Integrated into the trip initiation logic are two RPS Manual Trip channels. Manual Trip #1 operates by directly de-energizing all four M contactors in response to the operation of a manual pushbutton. Manual Trip #2 operates by de-energizing an undervoltage relay which results in the opening of two circuit breakers, CB-AB and CB-CD, which supply power to the CEDM clutch power supplies. Manual Trip channel #1 consists of manual trip pushbutton #1 and interconnecting wiring. Manual Trip channel #2 consists of manual trip pushbutton #2, circuit breakers CB-AB and CB-CD, and associated interconnecting wiring.

With one manual reactor trip channel inoperable, operation may continue until the reactor is shut down for other reasons. No safety analyses assume operation of the Manual trip. Because of this, the Required Action is to restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3 during the next plant startup.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The ESF logic system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968. Two functionally redundant ESF logic subsystems "A" and "B" are provided to ensure high reliability and effective in-service testing. These logic subsystems are designed for individual reliability and maximum attainable mutual independence both physically and electrically. Either logic subsystem acting alone can automatically actuate engineered safety features and essential supporting systems.

All Engineered Safety Features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-2 basis. The number of installed channels for Containment Radiation High Signal (CRHS) is two. CRHS isolates the containment pressure relief, air sample and purge system valves.

Entry into Technical Specification 2.15.1(3) is made when conditions have caused one logic subsystem ("A" or "B") to become inoperable but the redundant logic subsystem remains operable. The loss of a prime initiation relay (which renders all 4 channels of a logic subsystem inoperable) is the condition most likely to cause entry into Technical Specification 2.15.1(3). In this situation, the remaining ESF logic subsystem still has the capability to automatically actuate engineered safety features equipment and essential supporting systems. The 48-hour completion time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining ESF logic subsystem. Technical Specification 2.15.1(3) will not be used upon loss of the common channels that affect both "A" and "B" subsystems prime initiators operability unless the permissible bypass condition is met. Upon exiting TS 2.15.1(3) following the restoration of a prime initiation relay to OPERABLE status, if any channel(s) remain inoperable, the appropriate Limiting Conditions for Operation (LCO) (TS 2.15.1(1) or TS 2.15.1(2) is applicable with the length of inoperability measured from time of discovery of: 1) prime initiation relay inoperable, or 2) channel inoperability, whichever is longer.

The ESF system provides a 2-out-of-4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a LCO on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEAs to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an Anticipated Operational Occurrence (AOO) and factored into the derivation of the Limiting Safety System Settings (LSSS) and LCO. With the rod block function out-of-service, several additional CEA deviation events must be considered as AOOs. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the LCO and LSSS are valid for a Group 4 insertion of less than or equal to 15%.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

Operability of the primary CEA position indication system (CEAPIS) channel and the secondary CEAPIS channel is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits of TS 2.10.2. The primary CEAPIS channel utilizes the output of a synchro transmitter geared to the clutch output shaft. CEA position is displayed visually at the main control panel.

The secondary CEAPIS channel utilizes the output of a voltage divider network controlled by a series of reed switches. The reed switches are actuated by a permanent magnet attached to the rack assembly. Position information is supplied to the distributed control system (DCS) flat-panel touch monitors for simultaneous viewing of all CEA group positions.

Limit switches on the regulating CEAs and reed switches on the shutdown CEAs provide an additional means of determining CEA position when the CEAs are fully inserted or fully withdrawn. When the CEAs are fully inserted or fully withdrawn, this indication (displayed on the DCS) can be used in lieu of secondary CEAPIS data to meet the shiftly CHANNEL CHECK of primary CEAPIS. However, as limit switch indication is not fully independent of secondary CEAPIS, primary CEAPIS must be used to verify secondary CEAPIS data.

In MODES 1 and 2, CEA position indication is required to allow verification that the CEAs are positioned and aligned as assumed in the safety analysis. If one CEA position indication channel is inoperable for one or more CEAs, TS 3.1, Table 3-3, Item 4 (CEA position verification) must be performed within 15 minutes following any CEA motion in that group to ensure that the CEAs are positioned as required.

The operability of the Alternate Shutdown Panel (AI-185), including Wide Range Logarithmic Power and Source Range Monitors on AI-212, and Emergency Auxiliary Feedwater Panel (AI-179) instrument and control circuits ensures that sufficient capability is available to permit entry into and maintenance of the Hot Shutdown Mode from locations outside of the Control Room. This capability is required in the event that Control Room habitability is lost due to fire in the cable spreading room or Control Room.

Variances which may exist at startup between the more accurate ΔT-Power and Nuclear Instrumentation Power (NI-Power) are not significant for enabling of the trip functions. By 15% of rated power as measured by the uncalibrated NI Power, the Axial Power Distribution (APD) and Loss of Load (LOL) trip functions are enabled while the High Rate of Change of Power trip is bypassed.

The APD trip function acts to limit the axial power shape to the range assumed in the setpoint analysis. Significant margins to local power density limits exist at 15% power, as well as power levels up to at least 30% (where NI calibration occurs).

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The LOL trip function acts as an anticipatory trip for the high pressurizer pressure and high power trips in order to limit the severity of a LOL transient. This trip is not credited in the USAR Chapter 14 Safety Analyses and any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

The High Rate of Change of Power trip acts to limit power excursions from low power levels and bypassing of this trip at a high power level is conservative. This trip is not credited in the USAR Chapter 14 Safety Analyses for Mode 1 operation. Any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

Steam generator blowdown isolation ensures that the auxiliary feedwater system performs its design function of maintaining adequate steam generator (SG) water level for decay heat removal once the auxiliary feedwater actuation signal (AFAS) is actuated. The steam generator blowdown isolation function consists of two trains (logic subsystems). Each train closes one SG blowdown isolation valve to each SG. Each SG has redundant (Train A and Train B) blowdown isolation valves. Four clutch power relays initiate closure of the SG blowdown isolation valves with each clutch power relay closing one valve when the reactor trips. Failure of one clutch power relay to initiate SG blowdown isolation or failure of one train will not prevent single valve isolation of SG blowdown flow.

References

(1) USAR, Section 7.2.7.1

TABLE 2-2

Instrument Operating Requirements for Reactor Protective System

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	~~Manual (Trip Buttons)~~ Not Used	~~1~~	~~None~~	~~None~~	~~N/A~~
2	High Power Level	$2^{(b)(c)}$	$1^{(c)}$	Thermal Power Input Bypassed below 10^{-4}% of Rated Power$^{(a)(d)}$	(e)
3	Thermal Margin/Low Pressurizer Pressure	$2^{(b)}$	1	Below 10^{-4}% of Rated Power$^{(a)(d)}$	(e)
4	High Pressurizer Pressure	$2^{(b)}$	1	None	(e)
5	Low R.C. Flow	$2^{(b)}$	1	Below 10^{-4}% of Rated Power $^{(a)(d)}$	(e)
6	Low Steam Generator Water Level	2/Steam Gen$^{(b)}$	1/Steam Gen	None	(e)
7	Low Steam Generator Pressure	2/Steam Gen$^{(b)}$	1/Steam Gen	Below 600 psia$^{(a)(d)}$	(e)
8	Containment High Pressure	$2^{(b)}$	1	During Leak Test	(e)
9	Axial Power Distribution	$2^{(b)(c)}$	$1^{(c)}$	Below 15% of Rated Power$^{(g)}$	(e)
10	High Rate Trip-wide Range Log Channels	$2^{(b)}$	1	Below 10^{-4}% and above 15% of Rated Power$^{(a)(g)}$	(e)
11	Loss of Load	$2^{(b)}$	1	Below 15% of Rated Power$^{(g)}$	(e)
12	Steam Generator Differential Pressure	$2^{(b)}$	1	None	(e)

a. Bypass automatically removed.

b. Specification 2.15.1(2) is applicable.

TABLE 2-2
(Continued)

c. If two channels are inoperable, load shall be reduced to 70% or less of rated power.

d. For low power physics testing this trip may be bypassed up to 10^{-1}% of rated power.

e. Specification 2.15.1(1) is applicable.

f. Deleted.

g. For each channel, the same bistable automatically activates the Loss of Load and Axial Power Distribution (APD) trips and automatically bypasses the high rate trip at 15% of rated power. Only the APD trip is a Limiting Safety System Setting. Therefore, the bistable is set to actuate within the APD tolerance band.

TABLE 2-3

Instrument Operating Requirements for Engineered Safety Features

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Safety Injection				
A	Manual	1	None	None	N/A
B	High Containment Pressure				
	Logic Subsystem A	2(a)(d)(l)	1	During Leak Test	(f)
	Logic Subsystem B	2(a)(d)(l)	1		
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2(a)(d)(l)	1	Reactor Coolant Pressure Less Than 1700 psia(b)	(f)
	Logic Subsystem B	2(a)(d)(l)	1		
2	Containment Spray				
A	Manual(m)	1	None	None	N/A
B	High Containment Pressure				
	Logic Subsystem A	2(a)(c)(d)(l)	1	During Leak Test	(f)
	Logic Subsystem B	2(a)(c)(d)(l)	1		
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2(a)(c)(d)(l)	1	Reactor Coolant Pressure Less Than 1700 psia(b)	(f)
	Logic Subsystem B	2(a)(c)(d)(l)	1		
D	Steam Generator Low/Low Pressure				
	Logic Subsystem A	2/Steam Gen(a)(c)(d)(l)	1/Steam Gen	Steam Generator Pressure Less Than 600 psia(n)	(f)
	Logic Subsystem B	2/Steam Gen(a)(c)(d)(l)	1/Steam Gen		
3	Recirculation				
A	Manual	1	None	None	N/A
B	SIRW Tank Low Level				
	Logic Subsystem A	2(a)(k)(l)	1	None	(j)
	Logic Subsystem B	2(a)(k)(l)	1		
4	Emergency Off-Site Power Trip				
A	Manual	1(e)	None	None	N/A
B	Emergency Bus Low Voltage (Each Bus)				
	-Loss of Voltage	2(d)	1	Reactor Coolant Temperature Less Than 300° F	(f)
	-Degraded Voltage	2(a)(d)	1		

TABLE 2-3
(Continued)

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
5	Auxiliary Feedwater				
A	Manual	1	None	None	N/A
B	Auto. Initiation Logic Subsystem A Logic Subsystem B			Operating Modes 3, 4, and 5	
	-Steam Generator Low Level	2(a)(d)(l)	1		(h)
	-Steam Generator Low Pressure	3(a)(g)(l)	1		(i)
	-Steam Generator Differential Pressure	3(a)(g)(l)	1		(i)

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass above 1700 psia.

c Coincident containment high pressure, pressurizer low/low pressure, and steam generator low pressure signals are required for initiation of containment spray.

d If minimum OPERABLE channel conditions are reached, one inoperable channel must be placed in the tripped condition or low level actuation position for auxiliary feedwater system within eight hours from the time of discovery of loss of operability. Specification 2.15.1(2) is applicable.

e Control switch on incoming breaker.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from time of discovery of loss of operability. Specification 2.15.1(1) is applicable.

g Three channels required because bypass or failure results in auxiliary feedwater actuation block in the affected channel.

h Specification 2.15.1(1) is applicable.

TABLE 2-3
(Continued)

i If the channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If one of the four channels on one steam generator is in prolonged bypass and a channel on the other steam generator becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

j If one channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If a channel is in prolonged bypass and a channel on the opposite train becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

k Specification 2.15.1(2) is applicable.

l Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

m Steam Generator Low Pressure permissive is required for actuation.

n Auto removal of bypass prior to exceeding 600 psia.

TABLE 2-4

Instrument Operating Conditions for Isolation Functions

No.	**Functional Unit**	**Minimum Operable Channels**	**Minimum Degree of Redundancy**	**Permissible Bypass Condition**	**Test, Maintenance and Inoperable Bypass**
1	Containment Isolation				
A	Manual	1	None	None	N/A
B	Containment High Pressure Logic Subsystem A Logic Subsystem B	 2(a)(e)(g) 2(a)(e)(g)	 1 1	 During Leak Test	 (f)
C	Pressurizer Low/Low Pressure Logic Subsystem A Logic Subsystem B	 2(a)(e)(g) 2(a)(e)(g)	 1 1	 Reactor Coolant Pressure Less Than 1700 psia(b)	 (f)
2	Steam Generator Isolation				
A	Manual	1	None	None	N/A
B	Steam Generator Isolation	 1	 None	 None	 N/A
	(i) Steam Generator Low Pressure Logic Subsystem A	 2/Steam Gen(a)(e)(g)	 1/Steam Gen	 Steam Generator Pressure Less Than 600 psia(c)	 (f)
	Logic Subsystem B	2/Steam Gen(a)(e)(g)	1/Steam Gen		
	(ii) Containment High Pressure Logic Subsystem A Logic Subsystem B	 2(a)(e)(g) 2(a)(e)(g)	 1 1	 During Leak Test	 (f)
3	Ventilation Isolation				
A	Manual	1	None	None	N/A
B	Containment High Radiation Logic Subsystem A Logic Subsystem B	 1(d)(g) 1(d)(g)	 None None	 If Containment Relief and Purge Valves are Closed	 (f)
4	Steam Generator Blowdown Isolation				
A	Manual	1(h)	None	Operating Modes 3, 4, & 5	N/A
B	Reactor Trip Trains A and B	2(h)(i)	None	Operating Modes 3, 4, & 5 OR if at least one valve for each steam generator is closed	(j)

TABLE 2-4
(Continued)

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass prior to exceeding 1700 psia.

c Auto removal of bypass prior to exceeding 600 psia.

d A and B trains are both actuated by either the Containment or Auxiliary Building Exhaust Stack initiating channels. The number of installed channels for Containment Radiation High Signal is two for purposes of Specification 2.15.1(1).

e If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within eight hours from the time of discovery of loss of operability. Specification 2.15.1(2) is applicable.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from the time of discovery of loss of operability. Specification 2.15.1(1) is applicable.

g Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

h. "Minimum Operable Channels" for steam generator blowdown isolation refers to the minimum number of trains (logic subsystems) which are required to be operable to provide manual or automatic SG blowdown isolation.

i. If both trains become inoperable, power operation may continue provided at least one SG blowdown isolation valve for each steam generator is closed OR be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15.1(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

j. If one train becomes inoperable, that train may be placed in the bypassed condition. If the train is not returned to OPERABLE status within 24 hours from time of discovery of loss of operability, operation may continue as long as one SG blowdown isolation valve to each steam generator is closed. If the train is not returned to OPERABLE status within 24 hours from time of discovery, with blowdown not isolated to both SGs, be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15.1(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

TABLE 2-5

Instrumentation Operating Requirements for Other Safety Feature Functions

No.	**Functional Unit**	**Minimum Operable Channels**	**Minimum Degree of Redundancy**	**Permissible Bypass Condition**
1	CEA Position Indication Systems	1[(a)]	None	None
2	Pressurizer Level	1	None	Not Applicable

NOTES:

(a) If one channel of CEA position indication is inoperable for one or more CEAs, requirements of specification 2.15.1 are modified for item 1 to "Perform TS 3.1, Table 3-3. Item 4 within 15 minutes following any CEA motion in that group." Specifications 2.15.1(1), (2), and (3) are not applicable.

TABLE 2-6

Alternate Shutdown and Auxiliary Feedwater Panel Functions

Function/Instrument or Control Parameter	Location	Required Number of Channels
1. Reactivity Control		
a. Source Range Power	AI-212	1
b. Reactor Wide Range Logarithmic Power	AI-212	1
2. Reactor Coolant System Pressure Control		
a. Pressurizer Wide Range Pressure (0-2500 psia)	AI-179	1
3. Decay Heat Removal via Steam Generators		
a. Reactor Coolant Hot Leg Temperature	AI-185	1 (Note 1)
b. Reactor Coolant Cold Leg Temperature	AI-185	1 (Note 1)
c. Steam Generator Pressure	AI-179	1 per Steam Generator
d. Steam Generator Narrow Range Level	AI-179	1 per Steam Generator
e. Steam Generator Wide Range Level	AI-179	1 per Steam Generator
4. Reactor Coolant System Inventory Controls		
a. Pressurizer Level	AI-185	1
b. Volume Control Tank Level	AI-185	1
c. Charging Pump CH-1B and its associated controls	AI-185	1
d. Charging Isolation Valve Control	AI-185	1
5. Transfer Functions		
a. All Transfer Switches/Lockout Relays	AI-185	1
b. All Transfer Switches/Lockout Relays	AI-179	1

Note 1: One reactor coolant hot leg temperature indication and one reactor coolant cold leg temperature indication channel must both be operable on the same steam generator (i.e., RC-2A or RC-2B).

TABLE 2-6
(Continued)

Alternate Shutdown and Auxiliary Feedwater Panel Functions

Function/Instrument or Control Parameter	Location	Required Number of Channels
6. Auxiliary Feedwater Controls		
a. Steam Generator RC-2A and 2B Auxiliary Feedwater Isolation Inboard and Outboard Valves Control	AI-179	1
b. Steam-Driven Pump FW-10 Recirculation Valve Control	AI-179	1
c. Steam-Driven Pump FW-10 Steam Isolation Valve Control	AI-179	1
d. Steam from Steam Generator RC-2A and RC-2B to FW-10 Steam Isolation Valve Control	AI-179	1

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TABLE OF CONTENTS (ALPHABETICAL ORDER)

2.0 LIMITING CONDITIONS FOR OPERATION

2.15.1 Instrumentation and Control Systems

Applicability

Applies to plant instrumentation systems.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specifications

The operability, permissible bypass, and Test Maintenance and Inoperable bypass specifications of the plant instrument and control systems shall be in accordance with Tables 2-2 through 2-5.

(1) In the event the number of channels of a particular system in service falls one below the total number of installed channels, the inoperable channel shall be placed in either the bypassed or tripped condition within one hour if the channel is equipped with a bypass switch, and eight hours if jumpers or blocks must be installed in the control circuitry. The inoperable channel may be bypassed for up to 48 hours from time of discovering loss of operability; however, if the inoperability is determined to be the result of malfunctioning RTDs or nuclear detectors supplying signals to the high power level, thermal margin/low pressurizer pressure, and axial power distribution channels, these channels may be bypassed for up to 7 days from time of discovering loss of operability. If the inoperable channel is not restored to OPERABLE status after the allowable time for bypass, it shall be placed in the tripped position or, in the case of malfunctioning RTDs or linear power nuclear detectors, the reactor shall be placed in hot shutdown within 12 hours. If active maintenance and/or surveillance testing is being performed to return a channel to active service or to establish operability, the channel may be bypassed during the period of active maintenance and/or surveillance testing. This specification applies to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(2) In the event the number of channels of a particular system in service falls to the limits given in the column entitled "Minimum Operable Channels," one of the inoperable channels must be placed in the tripped position or low level actuation permissive position for the auxiliary feedwater system within one hour, if the channel is equipped with a bypass switch, and within eight hours if jumpers or blocks are required; however, if minimum operable channel conditions for SIRW tank low signal are reached, both inoperable channels must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If at least one inoperable channel has not been restored to OPERABLE status after 48 hours from time of discovering loss of operability, the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the containment ventilation isolation valves are closed.

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~~249,~~ 263

2.0 LIMITING CONDITIONS FOR OPERATION

2.15.1 Instrumentation and Control Systems (Continued)

If after 24 hours from time of initiating a hot shutdown procedure at least one inoperable engineered safety features or isolation functions channel has not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. This specification applied to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(3) In the event the number of channels on a particular engineered safety features (ESF) or isolation logic subsystem in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," sufficient channels shall be restored to OPERABLE status within 48 hours so as to meet the minimum limits or the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure sufficient channels have not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours.

(4) In the event the number of channels of those particular systems in service not described in (3) above falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of OPERABLE high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above 10^{-4}% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

If during power operation, the rod block function of the secondary CEA position indication system and rod block circuit are inoperable for more than 24 hours, or the plant computer PDIL alarm, CEA group deviation alarm and the CEA sequencing function are inoperable for more than 48 hours, the CEAs shall be withdrawn and maintained at fully withdrawn and the control rod drive system mode switch shall be maintained in the off position except when manual motion of CEA Group 4 is required to control axial power distribution.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems
2.15.2 Reactor Protective System (RPS) Logic and Trip Initiation

Applicability

Applies to the operational status of RPS Logic and Trip Initiation channels in MODES 1 and 2; and,

When reactor coolant temperature (T_{cold}) is greater than 210°F or MODE 4 with more than one CEA capable of being withdrawn and RCS boron concentration less than REFUELING BORON CONCENTRATION.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specification

Six channels of RPS Logic matrices, four channels of RPS Trip Initiation Logic and two channels of RPS Manual Trip shall be OPERABLE.

Required Actions

(1) With one RPS Logic Matrix channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours.

(2) With one RPS Trip Initiation Logic channel inoperable, de-energize the affected clutch power supply within one hour.

(3) With one RPS Manual Trip channel inoperable, restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3.

(4) With two RPS Trip Initiation Logic channels affecting the same trip leg inoperable, de-energize the affected clutch power supplies immediately.

(5) With the required actions of (1), (2), or (4) not met, or
with two RPS Manual Trip channels inoperable, or
with two or more RPS Logic Matrices inoperable, or
with two or more RPS Trip Initiation Logic channels inoperable for reasons other than (4):

a. be in HOT SHUTDOWN and verify no more than one CEA is capable of being withdrawn within 6 hours,

OR

b. verify reactor coolant boron concentration is at REFUELING BORON CONCENTRATION within 6 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems
2.15.3 Alternate Shutdown and Auxiliary Feedwater Panel

Applicability

Applies to the operational status of Alternate Shutdown and Auxiliary Feedwater Panel Functions in MODES 1 and 2.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specification

The Alternate Shutdown and Auxiliary Feedwater Panel Functions/Instrumentation or Control Parameters in Table 2-6 shall be OPERABLE.

Required Actions

(1) With the number of OPERABLE channels or control circuits less than the required number of channels, restore the required number of channels to OPERABLE within seven (7) days.

(2) With the required actions of (1) not met, be in HOT SHUTDOWN within 12 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.15 Instrumentation and Control Systems (Continued)

Basis

During plant operation, the complete instrumentation systems will normally be in service. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protective system (RPS) and engineered safety features (ESF) system when one or more of the channels are out of service. Reactor safety is provided by RPS, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continued operation with certain instrumentation channels out of service since provisions were made for this in the plant design.

The RPS and most engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in the ESF logic system.

When one of the four channels is taken out of service for maintenance, RPS logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),(1) which results in a one-out-of-three channel logic. If in the 2-out-of-4 logic system of the RPS one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1-out-of-2. At rated power, the minimum OPERABLE high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are OPERABLE, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

An RPS Logic matrix channel consists of two matrix power supplies, four matrix relays and their associated contacts as well as all interconnecting wiring. An RPS Trip Initiation Logic channel consists of an M contactor and associated contacts, an interposing relay and all interconnecting wiring. Two RPS Trip Initiation Logic channels associated with the same pair of CEDM clutch power supplies are considered to affect the same trip leg.

Integrated into the trip initiation logic are two RPS Manual Trip channels. Manual Trip #1 operates by directly de-energizing all four M contactors in response to the operation of a manual pushbutton. Manual Trip #2 operates by de-energizing an undervoltage relay which results in the opening of two circuit breakers, CB-AB and CB-CD, which supply power to the CEDM clutch power supplies. Manual Trip channel #1 consists of manual trip pushbutton #1 and interconnecting wiring. Manual Trip channel #2 consists of manual trip pushbutton #2, circuit breakers CB-AB and CB-CD, and associated interconnecting wiring.

With one manual reactor trip channel inoperable, operation may continue until the reactor is shut down for other reasons. No safety analyses assume operation of the Manual trip. Because of this, the Required Action is to restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3 during the next plant startup.

2.0 LIMITING CONDITIONS FOR OPERATION
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The ESF logic system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968. Two functionally redundant ESF logic subsystems "A" and "B" are provided to ensure high reliability and effective in-service testing. These logic subsystems are designed for individual reliability and maximum attainable mutual independence both physically and electrically. Either logic subsystem acting alone can automatically actuate engineered safety features and essential supporting systems.

All Engineered Safety Features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-2 basis. The number of installed channels for Containment Radiation High Signal (CRHS) is two. CRHS isolates the containment pressure relief, air sample and purge system valves.

Entry into Technical Specification 2.15.1(3) is made when conditions have caused one logic subsystem ("A" or "B") to become inoperable but the redundant logic subsystem remains operable. The loss of a prime initiation relay (which renders all 4 channels of a logic subsystem inoperable) is the condition most likely to cause entry into Technical Specification 2.15.1(3). In this situation, the remaining ESF logic subsystem still has the capability to automatically actuate engineered safety features equipment and essential supporting systems. The 48-hour completion time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining ESF logic subsystem. Technical Specification 2.15.1(3) will not be used upon loss of the common channels that affect both "A" and "B" subsystems prime initiators operability unless the permissible bypass condition is met. Upon exiting TS 2.15.1(3) following the restoration of a prime initiation relay to OPERABLE status, if any channel(s) remain inoperable, the appropriate Limiting Conditions for Operation (LCO) (TS 2.15.1(1) or TS 2.15.1(2) is applicable with the length of inoperability measured from time of discovery of: 1) prime initiation relay inoperable, or 2) channel inoperability, whichever is longer.

The ESF system provides a 2-out-of-4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a LCO on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEAs to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an Anticipated Operational Occurrence (AOO) and factored into the derivation of the Limiting Safety System Settings (LSSS) and LCO. With the rod block function out-of-service, several additional CEA deviation events must be considered as AOOs. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the LCO and LSSS are valid for a Group 4 insertion of less than or equal to 15%.

2.0 LIMITING CONDITIONS FOR OPERATION
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

Operability of the primary CEA position indication system (CEAPIS) channel and the secondary CEAPIS channel is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits of TS 2.10.2. The primary CEAPIS channel utilizes the output of a synchro transmitter geared to the clutch output shaft. CEA position is displayed visually at the main control panel.

The secondary CEAPIS channel utilizes the output of a voltage divider network controlled by a series of reed switches. The reed switches are actuated by a permanent magnet attached to the rack assembly. Position information is supplied to the distributed control system (DCS) flat-panel touch monitors for simultaneous viewing of all CEA group positions.

Limit switches on the regulating CEAs and reed switches on the shutdown CEAs provide an additional means of determining CEA position when the CEAs are fully inserted or fully withdrawn. When the CEAs are fully inserted or fully withdrawn, this indication (displayed on the DCS) can be used in lieu of secondary CEAPIS data to meet the shiftly CHANNEL CHECK of primary CEAPIS. However, as limit switch indication is not fully independent of secondary CEAPIS, primary CEAPIS must be used to verify secondary CEAPIS data.

In MODES 1 and 2, CEA position indication is required to allow verification that the CEAs are positioned and aligned as assumed in the safety analysis. If one CEA position indication channel is inoperable for one or more CEAs, TS 3.1, Table 3-3, Item 4 (CEA position verification) must be performed within 15 minutes following any CEA motion in that group to ensure that the CEAs are positioned as required.

The operability of the Alternate Shutdown Panel (AI-185), including Wide Range Logarithmic Power and Source Range Monitors on AI-212, and Emergency Auxiliary Feedwater Panel (AI-179) instrument and control circuits ensures that sufficient capability is available to permit entry into and maintenance of the Hot Shutdown Mode from locations outside of the Control Room. This capability is required in the event that Control Room habitability is lost due to fire in the cable spreading room or Control Room.

Variances which may exist at startup between the more accurate ΔT-Power and Nuclear Instrumentation Power (NI-Power) are not significant for enabling of the trip functions. By 15% of rated power as measured by the uncalibrated NI Power, the Axial Power Distribution (APD) and Loss of Load (LOL) trip functions are enabled while the High Rate of Change of Power trip is bypassed.

The APD trip function acts to limit the axial power shape to the range assumed in the setpoint analysis. Significant margins to local power density limits exist at 15% power, as well as power levels up to at least 30% (where NI calibration occurs).

2.0 LIMITING CONDITIONS FOR OPERATION
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The LOL trip function acts as an anticipatory trip for the high pressurizer pressure and high power trips in order to limit the severity of a LOL transient. This trip is not credited in the USAR Chapter 14 Safety Analyses and any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

The High Rate of Change of Power trip acts to limit power excursions from low power levels and bypassing of this trip at a high power level is conservative. This trip is not credited in the USAR Chapter 14 Safety Analyses for Mode 1 operation. Any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

Steam generator blowdown isolation ensures that the auxiliary feedwater system performs its design function of maintaining adequate steam generator (SG) water level for decay heat removal once the auxiliary feedwater actuation signal (AFAS) is actuated. The steam generator blowdown isolation function consists of two trains (logic subsystems). Each train closes one SG blowdown isolation valve to each SG. Each SG has redundant (Train A and Train B) blowdown isolation valves. Four clutch power relays initiate closure of the SG blowdown isolation valves with each clutch power relay closing one valve when the reactor trips. Failure of one clutch power relay to initiate SG blowdown isolation or failure of one train will not prevent single valve isolation of SG blowdown flow.

References

(1) USAR, Section 7.2.7.1

TABLE 2-2

Instrument Operating Requirements for Reactor Protective System

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Not Used				
2	High Power Level	2(b)(c)	1(c)	Thermal Power Input Bypassed below 10^{-4}% of Rated Power(a)(d)	(e)
3	Thermal Margin/Low Pressurizer Pressure	2(b)	1	Below 10^{-4}% of Rated Power(a)(d)	(e)
4	High Pressurizer Pressure	2(b)	1	None	(e)
5	Low R.C. Flow	2(b)	1	Below 10^{-4}% of Rated Power (a)(d)	(e)
6	Low Steam Generator Water Level	2/Steam Gen(b)	1/Steam Gen	None	(e)
7	Low Steam Generator Pressure	2/Steam Gen(b)	1/Steam Gen	Below 600 psia(a)(d)	(e)
8	Containment High Pressure	2(b)	1	During Leak Test	(e)
9	Axial Power Distribution	2(b)(c)	1(c)	Below 15% of Rated Power(g)	(e)
10	High Rate Trip-wide Range Log Channels	2(b)	1	Below 10^{-4}% and above 15% of Rated Power(a)(g)	(e)
11	Loss of Load	2(b)	1	Below 15% of Rated Power(g)	(e)
12	Steam Generator Differential Pressure	2(b)	1	None	(e)

a. Bypass automatically removed.

b. Specification 2.15.1(2) is applicable.

TABLE 2-2
(Continued)

c. If two channels are inoperable, load shall be reduced to 70% or less of rated power.

d. For low power physics testing this trip may be bypassed up to 10^{-1}% of rated power.

e. Specification 2.15.1(1) is applicable.

f. Deleted.

g. For each channel, the same bistable automatically activates the Loss of Load and Axial Power Distribution (APD) trips and automatically bypasses the high rate trip at 15% of rated power. Only the APD trip is a Limiting Safety System Setting. Therefore, the bistable is set to actuate within the APD tolerance band.

TABLE 2-3

Instrument Operating Requirements for Engineered Safety Features

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Safety Injection				
A	Manual	1	None	None	N/A
B	High Containment Pressure Logic Subsystem A Logic Subsystem B	 2(a)(d)(l) 2(a)(d)(l)	 1 1	 During Leak Test	 (f)
C	Pressurizer Low/Low Pressure Logic Subsystem A Logic Subsystem B	 2(a)(d)(l) 2(a)(d)(l)	 1 1	 Reactor Coolant Pressure Less Than 1700 psia(b)	 (f)
2	Containment Spray				
A	Manual(m)	1	None	None	N/A
B	High Containment Pressure Logic Subsystem A Logic Subsystem B	 2(a)(c)(d)(l) 2(a)(c)(d)(l)	 1 1	 During Leak Test	 (f)
C	Pressurizer Low/Low Pressure Logic Subsystem A Logic Subsystem B	 2(a)(c)(d)(l) 2(a)(c)(d)(l)	 1 1	 Reactor Coolant Pressure Less Than 1700 psia(b)	 (f)
D	Steam Generator Low/Low Pressure Logic Subsystem A Logic Subsystem B	 2/Steam Gen(a)(c)(d)(l) 2/Steam Gen(a)(c)(d)(l)	 1/Steam Gen 1/Steam Gen	 Steam Generator Pressure Less Than 600 psia(n)	 (f)
3	Recirculation				
A	Manual	1	None	None	N/A
B	SIRW Tank Low Level Logic Subsystem A Logic Subsystem B	 2(a)(k)(l) 2(a)(k)(l)	 1 1	 None	 (j)
4	Emergency Off-Site Power Trip				
A	Manual	1(e)	None	None	N/A
B	Emergency Bus Low Voltage (Each Bus) -Loss of Voltage -Degraded Voltage	 2(d) 2(a)(d)	 1 1	 Reactor Coolant Temperature Less Than 300° F	 (f)

TABLE 2-3
(Continued)

No.	**Functional Unit**	**Minimum Operable Channels**	**Minimum Degree of Redundancy**	**Permissible Bypass Condition**	**Test, Maintenance and Inoperable Bypass**
5	Auxiliary Feedwater				
A	Manual	1	None	None	N/A
B	Auto. Initiation Logic Subsystem A Logic Subsystem B			Operating Modes 3, 4, and 5	
	-Steam Generator Low Level	2[(a)(d)(l)]	1		(h)
	-Steam Generator Low Pressure	3[(a)(g)(l)]	1		(i)
	-Steam Generator Differential Pressure	3[(a)(g)(l)]	1		(i)

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass above 1700 psia.

c Coincident containment high pressure, pressurizer low/low pressure, and steam generator low pressure signals are required for initiation of containment spray.

d If minimum OPERABLE channel conditions are reached, one inoperable channel must be placed in the tripped condition or low level actuation position for auxiliary feedwater system within eight hours from the time of discovery of loss of operability. Specification 2.15.1(2) is applicable.

e Control switch on incoming breaker.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from time of discovery of loss of operability. Specification 2.15.1(1) is applicable.

g Three channels required because bypass or failure results in auxiliary feedwater actuation block in the affected channel.

h Specification 2.15.1(1) is applicable.

TABLE 2-3
(Continued)

i If the channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If one of the four channels on one steam generator is in prolonged bypass and a channel on the other steam generator becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

j If one channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If a channel is in prolonged bypass and a channel on the opposite train becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

k Specification 2.15.1(2) is applicable.

l Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

m Steam Generator Low Pressure permissive is required for actuation.

n Auto removal of bypass prior to exceeding 600 psia.

TABLE 2-4

Instrument Operating Conditions for Isolation Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Containment Isolation				
A	Manual	1	None	None	N/A
B	Containment High Pressure				
	Logic Subsystem A	2(a)(e)(g)	1	During Leak Test	(f)
	Logic Subsystem B	2(a)(e)(g)	1		
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2(a)(e)(g)	1	Reactor Coolant Pressure Less Than 1700 psia(b)	(f)
	Logic Subsystem B	2(a)(e)(g)	1		
2	Steam Generator Isolation				
A	Manual	1	None	None	N/A
B	Steam Generator Isolation	1	None	None	N/A
	(i) Steam Generator Low Pressure				
	Logic Subsystem A	2/Steam Gen(a)(e)(g)	1/Steam Gen	Steam Generator Pressure Less Than 600 psia(c)	(f)
	Logic Subsystem B	2/Steam Gen(a)(e)(g)	1/Steam Gen		
	(ii) Containment High Pressure				
	Logic Subsystem A	2(a)(e)(g)	1	During Leak Test	(f)
	Logic Subsystem B	2(a)(e)(g)	1		
3	Ventilation Isolation				
A	Manual	1	None	None	N/A
B	Containment High Radiation				
	Logic Subsystem A	1(d)(g)	None	If Containment Relief and Purge Valves are Closed	(f)
	Logic Subsystem B	1(d)(g)	None		
4	Steam Generator Blowdown Isolation				
A	Manual	1(h)	None	Operating Modes 3, 4, & 5	N/A
B	Reactor Trip Trains A and B	2(h)(i)	None	Operating Modes 3, 4, & 5 OR if at least one valve for each steam generator is closed	(j)

TABLE 2-4
(Continued)

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass prior to exceeding 1700 psia.

c Auto removal of bypass prior to exceeding 600 psia.

d A and B trains are both actuated by either the Containment or Auxiliary Building Exhaust Stack initiating channels. The number of installed channels for Containment Radiation High Signal is two for purposes of Specification 2.15.1(1).

e If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within eight hours from the time of discovery of loss of operability. Specification 2.15.1(2) is applicable.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from the time of discovery of loss of operability. Specification 2.15.1(1) is applicable.

g Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

h. "Minimum Operable Channels" for steam generator blowdown isolation refers to the minimum number of trains (logic subsystems) which are required to be operable to provide manual or automatic SG blowdown isolation.

i. If both trains become inoperable, power operation may continue provided at least one SG blowdown isolation valve for each steam generator is closed OR be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15.1(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

j. If one train becomes inoperable, that train may be placed in the bypassed condition. If the train is not returned to OPERABLE status within 24 hours from time of discovery of loss of operability, operation may continue as long as one SG blowdown isolation valve to each steam generator is closed. If the train is not returned to OPERABLE status within 24 hours from time of discovery, with blowdown not isolated to both SGs, be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15.1(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

TABLE 2-5

Instrumentation Operating Requirements for Other Safety Feature Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition
1	CEA Position Indication Systems	1[a]	None	None
2	Pressurizer Level	1	None	Not Applicable

NOTES:

(a) If one channel of CEA position indication is inoperable for one or more CEAs, requirements of specification 2.15.1 are modified for item 1 to "Perform TS 3.1, Table 3-3. Item 4 within 15 minutes following any CEA motion in that group." Specifications 2.15.1(1), (2), and (3) are not applicable.

TABLE 2-6

Alternate Shutdown and Auxiliary Feedwater Panel Functions

Function/Instrument or Control Parameter	Location	Required Number of Channels
1. Reactivity Control		
a. Source Range Power	AI-212	1
b. Reactor Wide Range Logarithmic Power	AI-212	1
2. Reactor Coolant System Pressure Control		
a. Pressurizer Wide Range Pressure (0-2500 psia)	AI-179	1
3. Decay Heat Removal via Steam Generators		
a. Reactor Coolant Hot Leg Temperature	AI-185	1 (Note 1)
b. Reactor Coolant Cold Leg Temperature	AI-185	1 (Note 1)
c. Steam Generator Pressure	AI-179	1 per Steam Generator
d. Steam Generator Narrow Range Level	AI-179	1 per Steam Generator
e. Steam Generator Wide Range Level	AI-179	1 per Steam Generator
4. Reactor Coolant System Inventory Controls		
a. Pressurizer Level	AI-185	1
b. Volume Control Tank Level	AI-185	1
c. Charging Pump CH-1B and its associated controls	AI-185	1
d. Charging Isolation Valve Control	AI-185	1
5. Transfer Functions		
a. All Transfer Switches/Lockout Relays	AI-185	1
b. All Transfer Switches/Lockout Relays	AI-179	1

Note 1: One reactor coolant hot leg temperature indication and one reactor coolant cold leg temperature indication channel must both be operable on the same steam generator (i.e., RC-2A or RC-2B).

TABLE 2-6
(Continued)

Alternate Shutdown and Auxiliary Feedwater Panel Functions

Function/Instrument or Control Parameter	Location	Required Number of Channels
6. Auxiliary Feedwater Controls		
a. Steam Generator RC-2A and 2B Auxiliary Feedwater Isolation Inboard and Outboard Valves Control	AI-179	1
b. Steam-Driven Pump FW-10 Recirculation Valve Control	AI-179	1
c. Steam-Driven Pump FW-10 Steam Isolation Valve Control	AI-179	1
d. Steam from Steam Generator RC-2A and RC-2B to FW-10 Steam Isolation Valve Control	AI-179	1