PVNGS Technical Specification Bases (TS Bases)
Revision 62
Replacement Pages and Insertion Instructions

The following LDCRs are included in this change:

Technical Specification Bases Revision 62 includes the following changes:

- LDCR 13-B002 reflects conforming changes to the Technical Specification (TS) Bases to implement License Amendment 195, dated March 30, 2015. The changes involve the adoption of TSTF-486, *Revise MTC Surveillance for Startup Test Activity Reduction (STAR) Program (WCAP-17787)*. Specifically, Surveillance Requirements 3.1.4.1 and 3.1.4.2 were modified.

- LDCR 14-B008 reflects clarification of the TS Bases description of a dropped 4-fingered Control Element Assemble (CEA), consistent with the CRDR 4584784 Evaluation. Specifically, TS Bases 3.3.1, *Reactor Protective System (RPS) Instrumentation - Operating*, has been updated to reflect that the protection for the 4-finger CEA event is provided by installed thermal margin and operator actions to reduce power to meet the requirements of the *Core Operating Limits Report (COLR)*.

Insertion Instructions

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PVNGS

Palo Verde Nuclear Generating Station
Units 1, 2, and 3

Technical Specification Bases

Revision 62
April 15, 2015

PALO VERDE

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B 3.8.2-1	0	B 3.8.8-1	1
B 3.8.2-2	0	B 3.8.8-2	1
B 3.8.2-3	0	B 3.8.8-3	21
B 3.8.2-4	21	B 3.8.8-4	56
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups and temperatures during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers, either fixed lumped poison rods or poisons distributed within selected fuel rods to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding.

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control element assembly (CEA) withdrawal transient from either subcritical or full THERMAL POWER. The limiting overheating event relative to plant response is based on the Loss of Condenser Vacuum event (Ref. 3). The most limiting event with respect to a positive MTC is a CEA withdrawal accident from a subcritical or low (hot zero) power condition, also referred to as a startup accident (Ref. 4).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. A middle of cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

LCO 3.1.4 requires the MTC to be within the positive and negative limits specified in the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The positive MTC limit in the COLR ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

The MTC limit specified in the LCO is the maximum positive MTC value approved in the plant's licensing basis and ensures that the reactor operates with a negative MTC over the largest possible range fuel cycle operation. The cycle-specific MTC limit specified in the COLR must be equal to or less positive than the MTC limit specified in the LCO.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. Limited control of MTC can be achieved by adjusting CEA position and boron concentration. During operation, the LCO can be ensured through measurement and adjustments to CEA position and boron concentration. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure accidents, such as the uncontrolled CEA assembly or group withdrawal, will not violate the

(continued)

BASES (continued)

APPLICABILITY (continued)	assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES except for a MSLB in MODE 3. In this case, the analysis assumes worst case MTC, with the ECCS systems mitigating the event. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.
ACTIONS	A.1 MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.1.4.1 and SR 3.1.4.2 The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. For fuel cycles that meet the applicability requirements in Reference 5, and specifically the acceptance criteria that must be met in order to substitute the measured value of MTC at hot zero power (HZP) with an alternate MTC value, SR 3.1.4.1 may be met prior to entering MODE 1 after each fuel loading by confirmation that the predicted MTC, when

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2 (continued)

adjusted for the measured RCS boron concentration, is within the most positive (least negative) MTC limit specified in the COLR. If this adjusted predicted MTC value is used to meet the SR prior to entering MODE 1, a confirmation by measurement that MTC is within the upper MTC limit must be performed in MODE 1 within 7 Effective Full Power Days (EFPD) of reaching 40 EFPD of core burnup. The applicability requirements in Reference 5 ensure core designs are not significantly different from those used to benchmark predictions and require that the measured RCS boron concentration meets specific test criteria. This provides assurance that the MTC obtained from the adjusted predicted MTC is accurate.

For fuel cycles that do not meet the applicability requirements in Reference 5, the verification of MTC required prior to entering MODE 1 after each fuel loading is performed by calculation of the MTC based on measurement of the isothermal temperature coefficient. In this case, measurement of MTC within 7 EFPD of reaching 40 EFPD of core burnup is not required for SR 3.1.4.1.

The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value.

The requirement for measurement, within 7 EFPD of (before or after) reaching 40 EFPD and a $^{2}/_{3}$ core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the MTC limits specified in the COLR.

SR 3.1.4.2 is modified by a Note that indicates performance is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE prior to accomplishing the Surveillance is necessary.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2 (continued)

SR 3.1.4.2 is modified by a second Note, which indicates that if extrapolated MTC is more negative than the EOC limit specified in the COLR, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits. An extrapolation to the end of cycle is only required if the measurement at 2/3 cycle is performed.

SR 3.1.4.2 is modified by a third Note, which indicates that the Surveillance, which determines MTC 2/3 expected core burnup is only required if the MTC determined in SR 3.1.4.1 and 40 EFPD are not within 0.16*10E-4Δ k/k/°F of the corresponding design values. For cycles that meet the applicability requirements given in Reference 5, the MTC verification of MTC at 2/3 expected core burnup is not required if the result of the measurement at 40 EFPD is within a tolerance of 0.16*10E-4 Δk/k/°F of the corresponding design value.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. UFSAR, Section 15.0.
3. UFSAR, Section 15.2.
4. UFSAR, Section 15.4.
5. WCAP-17787, “Palo Verde Nuclear Generating Station STAR Program Implementation.”
6. CE-NPSD-911, “Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specification End-of-Cycle MTC Limit”, September 2000.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation – Operating

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during selected anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

Except for the Trip Function 6 and 7, the LSSS defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs). For Trip Functions 6 and 7, the UFSAR Trip Setpoint is the LSSS.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
- Fuel centerline melting shall not occur; and
- The Reactor Coolant System (RCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of

(continued)

BASES

BACKGROUND (continued)

occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- Bistable trip units;
- RPS Logic; and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal feature for those trips with operating bypasses. The RPS Logic and RTCBs are addressed in LCO 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation." The CEACs are addressed in LCO 3.3.3, "Control Element Assembly Calculators (CEACs)."

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The excore nuclear instrumentation, the core protection calculators (CPCs), and the CEACs, though complex, are considered components in the measurement channels of the Variable Over Power - High, Logarithmic Power Level - High, DNBR - Low, and Local Power Density (LPD) - High trips.

Four identical measurement channels, designated channels A through D, with electrical and physical separation, are provided for each parameter used in the generation of trip signals, with the exception of the control element assembly (CEA) position indication used in the CPCs. Each measurement channel provides input to one or more RPS bistables within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features Actuation System (ESFAS) bistables, and most provide indication in the control room.

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BASES

BACKGROUND

Reactor Trip Circuit Breakers (RTCBs) (continued)

Manual Trip circuitry includes the push button and interconnecting wiring to the RTCBs necessary to actuate both the undervoltage and shunt trip attachments but excludes the Initiation relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of individual RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. UFSAR, Section 7.2 (Ref. 8), explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Design Basis Definition

The RPS is designed to ensure that the following operational criteria are met:

- The associated actuation will occur when the parameter monitored by each channel reaches its setpoint and the specific coincidence logic is satisfied;
- Separation and redundancy are maintained to permit a channel to be out of service for testing or maintenance while still maintaining redundancy within the RPS instrumentation network.

Each of the analyzed accidents and transients (except for dropped 4-finger CEA event) can be detected by one or more RPS Functions. The accident analysis takes credit for most of the RPS trip Functions. Those functions for which no credit is taken, termed equipment protective functions, are not needed from a safety perspective.

Each RPS setpoint is chosen to be consistent with the function of the respective trip. The basis for each trip setpoint falls into one of three general categories:

Category 1: To ensure that the SLs are not exceeded during AOOs;

Category 2: To assist the ESFAS during accidents; and

Category 3: To prevent material damage to major plant components (equipment protective).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

The RPS maintains the SLs during selected AOOs and mitigates the consequences of DBAs in all MODES in which the RTCBs are closed.

Each of the analyzed transients and accidents can be detected by one or more RPS Functions. Functions not specifically credited in the accident analysis are part of the NRC staff approved licensing basis for the plant. Noncredited Functions include the Steam Generator #1 Level – High, and the Steam Generator #2 Level – High. These trips minimize the potential for equipment damage.

The specific safety analysis applicable to each protective function is identified below:

1. Variable Over Power-High (RPS)

   The Variable Over Power - High Trip (RPS-VOPT) is provided to protect the reactor core during positive reactivity addition excursions. Under steady state conditions the trip setpoint will stay above the neutron power level signal by a preset value, called the band function. When the power level increases the setpoint will increase to attempt to maintain the separation defined by the Band function, however the rate of the setpoint change is limited by the rate function. If the power level signal increases faster than the setpoint, a trip will occur when the power level eventually equals the trip setpoint. The maximum value the setpoint can have is determined by the ceiling function.

   A positive reactivity excursion transient will be detected by one or more RPS Functions. The Variable Over Power-High trip (RPS-VOPT) can provide protection against core damage during the following events:

   - Uncontrolled CEA Withdrawal From Subcritical and Low Power (AOO); and
   - CEA Ejection (Accident).

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BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

8, 9. Steam Generator Level – Low

The Steam Generator #1 Level – Low and Steam Generator #2 Level – Low trips ensure that a reactor trip signal is generated for the following events to help prevent exceeding the design pressure of the RCS due to the loss of the heat sink:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (AOO);
- Loss of Condenser Vacuum (AOO);
- Loss of Normal Feedwater Event (AOO);
- Feedwater System Pipe Break (Accident); and
- Single RCP Rotor Seizure (AOO)

10, 11. Steam Generator Level – High

The Steam Generator #1 Level – High and Steam Generator #2 Level – High trips are provided to protect the turbine from excessive moisture carryover in case of a steam generator overfill event. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.

12, 13. Reactor Coolant Flow – Low

The Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator #2-Low trips provide protection against an RCP Sheared Shaft Event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays below the pressure differential by a preset value called the step function, unless limited by a preset maximum decreasing rate determined by the Ramp Function, or a set minimum value determined by the Floor Function. The setpoints ensure that a reactor trip occurs to limit fuel failure and ensure offsite doses are within 10 CFR 100 guidelines.

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BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

14. Local Power Density – High

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The LPD – High trip provides protection against fuel centerline melting due to the occurrence of excessive local power density peaks during the following AOOs:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to the steam line rupture) Without Turbine Trip;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power; and
- CEA Misoperation, except for dropped 4-finger CEA event.

For the events listed above (except CEA Misoperation where the DNBR and LPD trips will occur near simultaneously), DNBR – Low will trip the reactor first, since DNB would occur before fuel centerline melting would occur.

Note that the protection for the 4-finger CEA event is provided by installed thermal margin and operator actions to reduce power to meet requirements of the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The DNBR – Low trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events and is the primary reactor trip (trips the reactor first) for these events:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip;
- Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component;
- Steam Line Break With Concurrent Loss of Offsite AC Power;
- Loss of Normal AC Power;
- Partial Loss of Forced Reactor Coolant Flow;
- Total Loss of Forced Reactor Coolant Flow;
- Single Reactor Coolant Pump (RCP) Shaft Seizure;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power;
- CEA Misoperation, except for dropped 4-finger CEA event;
- Primary Sample or Instrument Line Break; and
- Steam Generator Tube Rupture.

In the above list, only the steam line break, the steam generator tube rupture, the RCP shaft seizure, and the sample or instrument line break are accidents. The rest are AOOs.

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BASES

APPLICABLE SAFETY ANALYSES

15. Departure from Nucleate Boiling Ratio (DNBR)-Low (continued)

Note that the protection for the 4-finger CEA event is provided by installed thermal margin and operator actions to reduce power to meet requirements of the COLR.

In the safety analyses for transients involving reactivity and power distribution anomalies, credit may be taken for the CPC VOPT auxiliary trip algorithm in lieu of the RPS VOPT trip function. The exact trip credited (CPC or RPS) is documented in chapter 15 of the UFSAR under the individual event sections. The CPC VOPT auxiliary trip acts through the CPC DNBR-Low and LPD-High trip contacts to provide over power protection. When credit is taken for the CPC VOPT algorithm, the CPC VOPT setpoints installed in the plant are based on the safety analyses and may differ from the RPS VOPT allowable values and nominal setpoints. The setpoints associated with the CPC VOPT are controlled via Addressable Constants (TS Section 5.4.1) and Reload Data Block Constants (Ref. 8 and 13). The CPC VOPT auxiliary trip algorithm may provide protection against core damage during the following events:

- Uncontrolled CEA Withdrawal From Low Power (AOO);
- Uncontrolled CEA Withdrawal at Power (AOO);
- Single CEA Withdrawal within Deadband (AOO);
- Steam Bypass Control System Misoperation (AOO);
- CEA Ejection (Accident); and
- Main Steam Line Break (Accident).

(continued)

BASES

LCO

14. Local Power Density – High

This LCO requires four channels of LPD - High to be OPERABLE.

The LCO on the CPCs ensures that the SLs are maintained during selected AOOs and the consequences of accidents are acceptable.

A CPC is not considered inoperable if CEAC inputs to the CPC are inoperable. The Required Actions required in the event of CEAC channel failures ensure the CPCs are capable of performing their safety function.

The CPC channel has many redundant features designed to improve channel reliability. A minimum subset of features must be functional in order for the CPC to be capable of performing its safety related trip function. Therefore, the channel may remain OPERABLE in the presence of a subset of channel failures, while maintaining the ability to provide the LPD-High trip function.

On line CPC channel diagnostics make use of redundant features to maintain channel operability to the extent possible, and provide alarm and annunciation of detectable failures.

Those detectable CPC channel failures resulting in a loss of protective function and channel inoperability will result in a CPC Fail indication and associated Low DNBR and High LPD channel trips. Input failures resulting in a sensor out of range affecting one or more CPC process inputs will result in a CPC Sensor Failure indication. In addition, since the CPC software limits the sensor value to the lower or upper range limit value, a CPC channel trip would be generated in most cases due to these extreme values. Detectable failures, whether they result in a channel inoperability or not, are logged in a system event list.

Redundancy is demonstrated as follows:

a. Each CPC channel redundantly processes analog process and nuclear instrumentation inputs. Only one of the two redundant analog processing modules is required to maintain operability.

(continued)

BASES

LCO

14. Local Power Density – High (continued)

b. CEA position is redundantly processed by two CEA Position Processors (CPPs) in each CPC channel, and transmitted to the appropriate CEACs in all four CPC channels over one way fiber-optically isolated data links. Only one source of CEA position is required to maintain channel operability.

c. Each CPC channel has two redundant operator interface panels, a maintenance test panel (MTP) in the Core Protection Calculator System (CPCS) cabinet, and an Operator's Module (OM) in the control room. Neither is required for the CPC to perform its safety related function. However, one must be functional to assist personnel in performing certain surveillances. Upon failure of the OM, MTP, or both, the CPC channel will remain operable.

Each CPCS channel contains six processor modules. Failures of these modules are treated as follows:

- CPC Processor Module failure - this failure results in a CPC channel inoperability, as addressed by this LCO.
- Aux CPC Processor Module failure - this failure does not result in a CPC channel inoperability since this module does not perform any safety related functions.
- CEAC 1 Processor Module failure - this failure is addressed in LCO 3.3.3.
- CEAC 2 Processor Module failure - this failure is addressed in LCO 3.3.3.
- CPP 1 Processor Module failure - this failure is addressed in LCO 3.3.3.
- CPP 2 Processor Module failure - this failure is addressed in LCO 3.3.3.

(continued)

BASES

LCO

14. Local Power Density – High (continued)

The CPC channels may be manually bypassed below 1E-4% NRTP, as sensed by the logarithmic nuclear instrumentation. This bypass is enabled manually in all four CPC channels when plant conditions do not warrant the trip protection. The bypass effectively removes the DNBR – Low and LPD – High trips from the RPS Logic circuitry. The operating bypass is automatically removed when enabling bypass conditions are no longer satisfied.

The automatic bypass removal channel is INOPERABLE when the associated Log power channel has failed. The bypass function is manually controlled via station operating procedures and the bypass removal circuitry itself is fully capable of responding to a change in the associated input bistable. Footnotes (a) and (b) in Table 3.3.1-1 and (d) in Table 3.3.2-1 clearly require an "automatic" removal of trip bypasses. A failed Log channel may prevent, depending on the failure mode, the associated input bistable from changing state as power transitions through the automatic bypass removal setpoint. Specifically, when the indicated Log power channel is failed high (above 1E-4%), the automatic Hi-Log power trip bypass removal feature in that channel cannot function. Similarly, when the indicated Log power channel is failed low (below 1E-4%), the automatic DNBR-LPD trip bypass removal feature in that channel cannot function. Although one bypass removal feature is applicable above 1E-4% NRTP and the other is applicable below 1E-4% NRTP, both are affected by a failed Log power channel and should therefore be considered INOPERABLE.

(continued)

BASES

LCO 14. Local Power Density – High (continued)

When a Log channel is INOPERABLE, both the Hi-Log power and DNBR/LPD automatic trip bypass removal features in that channel are also INOPERABLE, requiring entry into LCO 3.3.1 Condition C or LCO 3.3.2 Condition C depending on plant operating MODE. Required Action C.1 for both LCOs 3.3.1 and 3.3.2 require the bypass channel to be disabled. Compliance with C.1 is met by placing the CR switches in "off" and "normal" for the Hi-Log power and DNBR/LPD bypasses respectively. No further action (key removal, periodic verification, etc.) is required. These CR switches are administratively controlled via station procedure therefore, the requirements of C.1 are continuously met.

This operating bypass is required to perform a plant startup, since both CPC generated trips will be in effect whenever shutdown CEAs are inserted. It also allows system tests at low power with Pressurizer Pressure - Low or RCPs off.

LCO 15. Departure from Nucleate Boiling Ratio (DNBR) – Low

This LCO requires four channels of DNBR - Low to be OPERABLE.

The LCO on the CPCs ensures that the SLs are maintained during selected AOOs and the consequences of accidents are acceptable.

A CPC is not considered inoperable if CEAC inputs to the CPC are inoperable. The Required Actions required in the event of CEAC channel failures ensure the CPCs are capable of performing their safety function.

The CPC channel has many redundant features designed to improve channel reliability. A minimum subset of features must be functional in order for the CPC to be capable of performing its safety related trip function. Therefore, the channel may remain OPERABLE in the presence of a subset of channel failures, while maintaining the ability to provide the DNBR-Low trip function. On line CPC channel diagnostics make use of redundant features to maintain channel operability to the extent possible, and provide alarm and annunciation of detectable failures.

(continued)

BASES

APPLICABILITY

Most RPS trips are required to be OPERABLE in MODES 1 and 2 because the reactor is critical in these MODES. The reactor trips are designed to take the reactor subcritical, which maintains the SLs during selected AOOs and assists the ESFAS in providing acceptable consequences during accidents. Most trips are not required to be OPERABLE in MODES 3, 4, and 5. In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM. Exceptions to this are:

- The Logarithmic Power Level – High trip, RPS Logic RTCBs, and Manual Trip are required in MODES 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events.

- Steam Generator Pressure-Low trip, is required in MODE 3, with the RTCBs closed to provide protection for steam line break events in MODE 3.

The Logarithmic Power Level – High trip, and the Steam Generator Pressure-Low trip in these lower MODES are addressed in LCO 3.3.2. The Logarithmic Power Level – High trip is bypassed prior to MODE 1 entry and is not required in MODE 1.

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

(continued)

BASES

ACTIONS

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, then all affected functions provided by that channel must be declared inoperable, and the unit must enter the Condition for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

One Note has been added to the ACTIONS. Note 1 has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit (Bypassed or Tripped)
1. Linear Power (Subchannel or Linear)	Variable Overpower (RPS) Local Power Density-High (RPS) DNBR-Low (RPS)
2. Pressurizer Pressure-High (Narrow Range)	Pressurizer Pressure-High (RPS) Local Power Density-High (RPS) DNBR-Low (RPS)
3. Steam Generator Pressure-Low	Steam Generator Pressure-Low (RPS) Steam Generator #1 Level-Low (ESF) Steam Generator #2 Level-Low (ESF)
4. Steam Generator Level-Low (Wide Range)	Steam Generator Level-Low (RPS) Steam Generator #1 Level-Low (ESF) Steam Generator #2 Level-Low (ESF)
5. Core Protection Calculator DNBR-Low (RPS)	Local Power Density-High (RPS)

(continued)