Bentley K. Jones
Director, Organizational Effectiveness
Harris Nuclear Plant
5413 Shearon Harris Rd
New Hill, NC 27562-9300

919.362.2305

August 10, 2016
Serial: HNP-16-064

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: Transmittal of Technical Specification Bases

Ladies and Gentlemen:

Pursuant to Technical Specification 6.8.4.n, Duke Energy Progress, Inc. hereby submits the Technical Specification Bases for the Shearon Harris Nuclear Power Plant, Unit 1. The entire Bases document is being submitted, which includes those changes implemented since the last update provided on May 18, 2015.

This submittal contains no regulatory commitments. Please refer any questions regarding this submittal to John Caves, Manager – Regulatory Affairs, at (919) 362-2406.

Sincerely,

Bentley K. Jones

Enclosure: Technical Specification Bases (102 pages)

cc: M. J. Riches, NRC Sr. Resident Inspector, HNP
M. Barillas, NRC Project Manager, HNP
NRC Regional Administrator, Region II

Bentley K. Jones
Director, Organizational Effectiveness
Harris Nuclear Plant
5413 Shearon Harris Rd
New Hill, NC 27562-9300

919.362.2305

August 10, 2016
Serial: HNP-16-064

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: Transmittal of Technical Specification Bases

Ladies and Gentlemen:

Pursuant to Technical Specification 6.8.4.n, Duke Energy Progress, Inc. hereby submits the Technical Specification Bases for the Shearon Harris Nuclear Power Plant, Unit 1. The entire Bases document is being submitted, which includes those changes implemented since the last update provided on May 18, 2015.

This submittal contains no regulatory commitments. Please refer any questions regarding this submittal to John Caves, Manager – Regulatory Affairs, at (919) 362-2406.

Sincerely,

Bentley K. Jones

Enclosure: Technical Specification Bases (102 pages)

cc: M. J. Riches, NRC Sr. Resident Inspector, HNP
M. Barillas, NRC Project Manager, HNP
NRC Regional Administrator, Region II

BASES
FOR
SECTION 2.0
SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (in this application, the HTP correlation for Siemens Fuel. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

2.1.1 REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}$, specified in the CORE OPERATING LIMITS REPORT (COLR) and a limiting axial power shape. An allowance is included for an increase in calculated $F_{\Delta H}$ at reduced power based on the expression:

$$F_{\Delta H} = F_{\Delta H}^{RTP} [1 + PF_{\Delta H} (1-P)]$$

Where P is the fraction of RATED THERMAL POWER,

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

$PF_{\Delta H}$ = Power Factor Multiplier for $F_{\Delta H}$ specified in the COLR.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f_1(\Delta I)$ function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer, and the RCS piping, pumps, valves and fittings are designed to Section III, Division I of the ASME Code for Nuclear Power Plants, which permits a maximum transient pressure of 110% to 125% of design pressure (2485 psig) depending on component. The Safety Limit of 2735 psig (110% of design pressure) is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. For example, if a bistable has a trip setpoint of 100%, a span of 125%, and a calibration accuracy of 0.5% of span, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is ≤ 100.62%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor and an increased rack drift factor, and provides a threshold value for determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Reactor Trip System Instrumentation Setpoints and TSTF-493

This section applies only to the Functional Units to which Notes 7 and 8 in the Trip Setpoint Column are applicable. Those Functional Units have revisions in accordance with Technical Specification Task Force Traveler 493 (TSTF-493). "Clarify Application of Setpoint Methodology for LSSS Functions." Those Functional Units are limited to

- Power Range, Neutron Flux High Setpoint
- Power Range, Neutron Flux Low Setpoint
- Power Range, Neutron Flux High Positive Rate, and
- Power Range, Neutron Flux High Negative Rate

Notes 7 and 8 have been added to Table 2.2-1 that require verifying both trip setpoint setting as-found and as-left values during surveillance testing. In accordance with 10 CFR 50.36, these functions are Limiting Safety System Settings. Adding test requirements ensures that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. These notes address NRC staff concerns with Technical Specification Allowable Values. Specifically, calculated Allowable Values may be non-conservative depending upon the evaluation of instrument performance history, and the as-left requirements of the calibration procedures could have an adverse effect on equipment

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

operability. In addition, using Allowable Values as the limiting setting for assessing instrument channel operability may not be fully in compliance with the intent of 10 CFR 50.36, and the existing surveillance requirements would not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. In the Harris Technical Specifications, the term Trip Setpoint is analogous to Nominal Trip Setpoint (NTSP) in TSTF-493.

Note 7 requires a channel performance evaluation when the as-found setting is outside its as-found tolerance. The performance evaluation verifies that the channel will continue to behave in accordance with safety analysis and instrument performance assumptions in the setpoint methodology. The purpose of this evaluation is to provide confidence in the performance prior to returning the channel to service. If the as-found setting is non-conservative with respect to the Allowable Value, the channel is INOPERABLE. If the as-found setting is conservative with respect to the Allowable Value but is outside the as-found tolerance band, the channel is OPERABLE but degraded. The degraded channel condition will be further evaluated during performance of the surveillance. This evaluation will consist of resetting the channel setpoint to within the as-left tolerances applicable to the actual setpoint implemented in the surveillance procedures (field setting), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition is entered into the corrective action program for further analysis and trending.

Note 8 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the Trip Setpoint in Table 2.2-1 or within as-left tolerances about a more conservative actual (field) setpoint. As-left channel settings outside the as-left tolerances of PLP-106 and the surveillance procedures cause the channel to be INOPERABLE.

A tolerance is necessary because no device perfectly measures the process. Additionally, it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The as-left tolerance is considered in the setpoint calculation. Failure to set the actual plant trip setpoint to within as-left the tolerances of the NTSP or within as-left tolerances of a more conservative actual field setpoint would invalidate the assumptions in the setpoint calculation, because any subsequent instrument drift would not start from the expected as-left setpoint. The determination will consider whether the instrument is degraded or is capable of being reset and performing its specified safety function. If the channel is determined to be functioning as required (i.e., the channel can be adjusted to within the as-left tolerance and is determined to be functioning normally based on the determination performed prior to returning the channel to service), then the channel is OPERABLE and can be restored to service. If the as-left instrument setting cannot be returned to a setting within the prescribed as-left tolerance band, the instrument would be declared INOPERABLE.

The methodologies for calculating the as-found tolerances and as-left tolerances about the Trip Setpoint or more conservative actual field setpoint are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints," which is incorporated by reference into the FSAR. The actual field setpoint and the associated as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report," the applicable section of which is incorporated by reference into the FSAR.

Limiting Trip Setpoint (LTSP) is generic terminology for the setpoint value calculated by means of the setpoint methodology documented in EGR-NGGC-0153. HNP uses the plant-specific term Nominal Trip Setpoint (NTSP) in place of the generic term LTSP. The NTSP is the LTSP with

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

margin added, and is always equal to or more conservative than the LTSP. The NTSP may use a setting value that is more conservative than the LTSP, but for Technical Specification compliance with 10 CFR 50.36, the plant-specific setpoint term NTSP is cited in Note 8.

The NTSP meets the definition of a Limiting Safety System Setting per 10 CFR 50.36 and is a predetermined setting for a protective channel chosen to ensure that automatic protective actions will prevent exceeding Safety Limits during normal operation and design basis anticipated operational occurrences, and assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Allowable Value is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual NTSP setting ensures that a Safety Limit is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance band, in accordance with uncertainty assumptions stated in the setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

Field setting is the term used for the actual setpoint implemented in the plant surveillance procedures, where margin has been added to the calculated field setting. The as-found and as-left tolerances apply to the field settings implemented in the surveillance procedures to confirm channel performance. A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the instrument operability must be verified based on the field setting and not the NTSP.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for transport to and response time of the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine inlet pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine inlet pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine inlet pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop; and below P-7, the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by the setpoint value. The Steam Generator Low Water level portion of the trip is activated when the setpoint value is reached, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine inlet pressure

Undervoltage and Underfrequency - Reactor Coolant Pump Buses (Continued)

at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine inlet pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS subsystems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3, if both the required Containment Spray Systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATION MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours. This specification is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

3.0.4 This specification establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher (LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2) MODE of

3.0.4 (Continued)

operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

3.0.5 This specification establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to 3.0.1 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

4.0.1 Specification 4.0.1 establishes the requirement that surveillances must be met during the OPERATIONAL MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual surveillances. This Specification is to ensure that surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a surveillance within the specified surveillance interval, in accordance with Specification 4.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire surveillance is performed within the specified surveillance interval. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated surveillances have been met. Nothing in the Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the surveillances; or

b. The requirements of the surveillance(s) are known not to be met between required surveillance performances.

Surveillances do not have to be performed when the unit is in an OPERATIONAL MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The surveillances associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given surveillance. In this case, the unplanned event may be credited as fulfilling the performance of the surveillance. This allowance includes those surveillances whose performance is normally precluded in a given OPERATIONAL MODE or other specified condition.

Surveillances, including surveillances invoked by ACTION requirements, do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specification 4.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with Specification 4.0.2. Post maintenance testing may not be possible in the current OPERATIONAL MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operations to proceed to an OPERATIONAL MODE or other specified condition where other necessary post maintenance tests can be completed.

4.0.1 (Continued)

An example of this process is Auxiliary Feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures that cannot be obtained until the unit is at HOT SHUTDOWN conditions. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

4.0.2 The provisions of this specification establish the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. Likewise, it is not the intent that the 18-month interval surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

4.0.3 Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified surveillance interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified surveillance interval was not met.

This delay period provides adequate time to complete surveillances that have been missed. This delay period permits the completion of a surveillance before complying with ACTION requirements or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

4.0.3 (Continued)

When a surveillance with a surveillance interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering OPERATIONAL MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified surveillance interval to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

Specification 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of OPERATIONAL MODE changes imposed by ACTION requirements.

Failure to comply with specified surveillance intervals for surveillance requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified surveillance interval is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the licensee's Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the time limits of the ACTION requirements for the applicable LCO begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable or the variable is outside the specified limits, and the time limits of the ACTION requirements for the applicable LCO begin immediately upon the failure of the surveillance.

Completion of the surveillance within the delay period allowed by this Specification, or within the completion time of the ACTIONS, restores compliance with Specification 4.0.1.

4.0.4 This specification establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

4.0.5 DELETED

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2 the most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1770 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4, and 5, the most restrictive condition occurs at BOL, when the boron concentration is the greatest. In these modes, the required SHUTDOWN MARGIN is composed of a constant requirement and a variable requirement, which is a function of the RCS boron concentration. The constant SHUTDOWN MARGIN requirement is based on an uncontrolled RCS cooldown from a steamline break accident, as is the case for MODES 1 and 2. The variable SHUTDOWN MARGIN requirement is based on the results of boron dilution accident analyses, where the SHUTDOWN MARGIN is varied as a function of RCS boron concentration, to guarantee a minimum of 15 minutes for operator action prior to a loss of SHUTDOWN MARGIN.

In modes 3, 4, and 5, the figure specified in the CORE OPERATING LIMITS REPORT (COLR) must be used with a curve giving the required shutdown boron concentrations for various temperatures as a function of core burnup. This cycle dependent relationship is provided for each cycle in the plant Curve Book. From the Curve Book, a required boron concentration that will provide adequate SHUTDOWN MARGIN can be determined and this concentration may be used to enter the figure specified in the COLR to determine the specific required SHUTDOWN MARGIN for that condition.

The boron dilution analysis assumed a common RCS volume and dilution flow rate for MODES 3 and 4, which differed from the volume and flow rate assumed for MODE 5 analysis. The MODE 5 conditions assumed limited mixing in the RCS and cooling with the RHR system only. In MODES 3 and 4, it was assumed that at least one reactor coolant pump was operating. If at least one reactor coolant pump is not operating in MODE 4, then the SHUTDOWN MARGIN requirements for MODE 5 shall apply, provided that the dilution flow rate assumed in the MODE 5 Boron Dilution analysis is not exceeded.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; i.e., the positive limit is based on core conditions for all rods withdrawn, BOL, hot zero THERMAL POWER, and the negative limit is based on core conditions for all rods withdrawn, EOL, RATED THERMAL POWER. Accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging/safety injection pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at BOL

BORATION SYSTEMS (Continued)

from full power equilibrium xenon conditions and requires 24,150 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 436,000 gallons of 2400-2600 ppm borated water be maintained in the refueling water storage tank (RWST).

With the RCS temperature below 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7150 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2400-2600 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified indicated levels used for surveillance testing, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

MOVABLE CONTROL ASSEMBLIES (Continued)

The intent of Technical Specification 3.1.3.1 ACTION statement "a" is to ensure, before leaving ACTION statement "a" and utilizing ACTION statement "c," that the rod urgent Failure alarm is illuminated or that an obvious electrical problem in the rod control system is detected by minimal electrical troubleshooting techniques. Expeditious action will be taken to determine if rod immovability is caused by an electrical problem in the rod control system.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the design DNBR value during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power;

$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power, with an allowance to account for measurement uncertainty.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference (target AFD) is determined at equilibrium xenon conditions. The rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target AFD at RATED THERMAL POWER for the associated core burnup conditions. Target AFD for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic measurement of the target flux difference value is necessary to reflect core burnup considerations. The target AFD may be updated between measurements based on the change in the predicted value with burnup.

AXIAL FLUX DIFFERENCE (Continued)

The target band about the target AFD is specified in the COLR. The target band limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits.

The computer determines the one-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the acceptable AFD target band. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

3/4.2.2 AND 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

$F_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. $F_{\Delta H}$ and $F_Q(Z)$ will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

FIGURE B 3/4 2-1 DELETED

HEAT FLUX HOT CHANNEL FACTOR NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained.

For the FSAR Chapter 15 analyses reliant on the Power Range Neutron Flux - High Trip Setting trip function, reduction of the Setpoint by approximately the same percentage as the required power reduction ensures DNBR margin is maintained.

When an $F_{\Delta H}$ measurement is taken, an allowance for measurement error must be applied prior to comparing to the $F_{\Delta H}^{RTP}$ limit(s) specified in the CORE OPERATING LIMITS REPORT (COLR). An allowance of 4% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. The margin is more than sufficient to offset any rod bow penalty and transition core penalty.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The hot channel factor $F_Q^M(Z)$ is measured periodically and increased by a cycle and height dependent power factor V(Z) to provide assurance that the

HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

limit on the hot channel factor, $F_Q(Z)$, is met. V(Z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The V(Z) function is specified in the COLR.

$F_Q^M(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to NRC should be submitted within 30 days in accordance with 10CFR50.4.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 594.8°F and 2185 psig, respectively, after an allowance for measurement uncertainty is included.

When RCS flow rate is measured, an additional allowance is necessary prior to comparison with the limit of Specification 3.2.5.c. Specifically for the precision calorimetric heat balance, a normal RCS flow rate error of 2.1% will be included.

Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance to 2.2% for no venturi fouling. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant parameters.

3/4.2.5 DNB PARAMETERS (Continued)

If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e. either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

Surveillance 4.2.5.1 ensures that temperature and pressure parameters, through instrument readout, are restored within their respective limits following load changes and other expected transient operation. The periodic surveillance of indicated RCS flow is intended to detect flow degradation.

Surveillance 4.2.5.2 allows entry into MODE 1, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. Measurement of RCS flow rate by performance of a precision calorimetric heat balance allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. The frequency of 18 months reflects the importance of verifying flow following a refueling outage, where work activities were performed that could affect RCS flow. Performance of a precision calorimetric at other times are unnecessary unless changes were introduced that would substantially reduce RCS flow and are likely to produce non-conservative results. The surveillance requirement to perform the precision calorimetric within 24 hours after exceeding 95% RTP is intended to stress the importance of collecting plant flow data as soon as practical after reaching a stable power level that is sufficient for performing the test and in recognition that some plants have experienced feedwater venturi fouling and other phenomena that are more probable as time elapses. If the precision calorimetric data can not be collected in the required time period, it is necessary to reduce power to less than 95% RTP until preparations are complete for collecting precision calorimetric data. Reducing power to less than 95%, resets the allowable time period requirement to perform the precision calorimeter within 24 hours after exceeding 95% RTP.

3/4.3.1 AND 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Trip System and Engineered Safety Features Actuation System instrumentation, and (3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report as approved by the NRC and documented in the SERs and SSER (letters to J. J. Sheppard from Cecil O. Thomas dated February 21, 1985; Roger A. Newton from Charles E. Rossi dated February 22, 1989; and Gerard T. Goering from Charles E. Rossi dated April 30, 1990).

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. For example, if a bistable has a trip setpoint of ≤100%, a span of 125%, and a calibration accuracy of ±0.50%, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is ≤100.62%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor draft factor, an increased rack drift factor, and provides a

threshold value for determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements; or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise or power interrupt tests); (2) inplace, onsite, or offsite (e.g., vendor) test measurements; or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Rev. 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP 14036-P-A, Rev. 1, "Elimination of Periodic Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component into operational service and re-verified following maintenance or modification that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for the repair are the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing element of a transmitter.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) charging/safety injection pumps start and automatic valves position, (2) reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment fan coolers start and automatic valves position, (11) emergency service water pumps start and automatic valves position, and (12) control room isolation and emergency filtration start.

Table 3.3-4 includes values for 6.9 kV Emergency Bus Undervoltage - Secondary (degraded grid) trip setpoints and allowable values. The secondary undervoltage relays are connected to two distinct time delay relays. Upon expiration of the first time delay, which is long enough to accommodate the starting of the motor which has the longest starting time, an alarm is actuated at the main control board to alert the operator of this condition and to permit operator actions to restore the system voltage. Automatic tripping actions as described for the primary protection are initiated if a safety actuation signal is present after the expiration of the time delay.

In the event of a coincident large break loss of coolant accident (LBLOCA) and voltage dropping to actuate the short-term DVR function (bus voltage drops into the range between the DVR dropout voltage setting and the loss of offsite power voltage setpoint), a safety injection actuation signal is generated, emergency loads begin to sequence onto the emergency buses (still powered from the normal offsite supply), and the emergency diesel generator starts but does not load. If the degraded voltage condition continues to exist until the short-term DVR time delay setting is reached, the emergency loads are then separated from offsite power, loads on emergency buses are shed, the emergency diesel generator output breaker is shut, and the emergency loads are sequenced back onto the emergency buses. The LBLOCA analysis timeline for the safety functions provided by the equipment in this scenario is used to establish the analytical limit for the maximum short-term DVR time delay. This meets the intent of Branch Technical Position PSB-1 regarding maximum time delays consistent with design basis accident analysis.

If degraded voltage conditions exist without a simultaneous accident (normal operating conditions), a longer time delay (Device 2-2) is allowed before the automatic tripping actions are initiated. This second time delay is based on the maximum time for which the most sensitive load can perform its safety function without impairment at the degraded voltage.

Calculations to determine time delay allowable values and trip setpoints to protect time delay analytical limits were performed consistent with the methodology of Technical Specification Task Force Traveler 493, Clarify Application of Setpoint Methodology for LSSS Functions. Although the DVR function is not a limiting safety system setting function, the methodology is a conservative approach for determination of these parameters.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and low steam-line pressure, sends an open signal to the accumulator discharge valves and automatically blocks steam-line isolation on a high rate of decrease in steam-line pressure. On decreasing pressurizer pressure, P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steam-line pressure and allows steam-line isolation, on a high rate of decrease in steam-line pressure, to become active upon manual block of Safety Injection from low steam-line pressure.

P-12 P-12 has no ESF or reactor trip functions. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of emergency systems.

3/4.3.3.2 DELETED

3/4.3.3.3 DELETED

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor.

This capability is consistent with General Design Criterion 3, 10 CFR 50.48(a) and 10 CFR 50.48(c).

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The RVLIS and In Core Thermocouple design meets the intent of Regulatory Guide 1.97. The HNP design (and Regulatory Guide 1.97) stipulates redundancy for RVLIS and In Core Thermocouples. A fully 100% functional channel would be available should a channel fail.

The RVLIS and In Core Thermocouple systems do not automatically actuate any component. These monitoring systems are used for indication only. Diverse monitoring is available for core cooling indication requirements such as Reactor Coolant Hot and Cold Leg temperature

indications as well as Reactor Coolant System pressure.

The thirty-day completion time for one inoperable channel of RVLIS or In Core Thermocouples is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring an instrument during this interval. If the thirty-day completion time was not met, then a written report to the NRC would be required to outline the preplanned alternate method of monitoring (in this case the other redundant channel would be available), the cause of the inoperability, and plans and a schedule for restoring the instrumentation channels of the Function to operable status.

If both channels of RVLIS or In Core Thermocouples are inoperable, then restore an inoperable channel within 7 days. The completion time of 7 days is based on the relatively low probability of an event requiring RVLIS and In Core Thermocouple instrumentation operation and the availability of alternate means to obtain the required information. Diverse monitoring is available for core cooling indication requirements such as Reactor Coolant Hot and Cold Leg temperature indications as well as Reactor Coolant System pressure. These parameters can be used to manually determine subcooling margin, which normally uses core exit temperatures.

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

3/4.3.3.9 DELETED

3/4.3.3.10 DELETED

3/4.3.3.11 DELETED

3/4.3.4 DELETED

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR value during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

Surveillance Requirements for MODES 3, 4, and 5 with reactor coolant loops filled require verification of steam generator (SG) OPERABILITY. Verification of adequate level in the applicable steam generator ensures an adequate heat sink for the removal of decay heat. If the SG tubes become uncovered, the associated loop may not be capable of providing the heat sink for the removal of the decay heat. The level values include allowances for channel uncertainty and process measurement effects and may not be simultaneously indicated by the respective instrumentation. The 12 hour frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 325°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any

SAFETY VALVES (Continued)

overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no reactor trip until the second Reactor Trip System trip setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water level in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water level also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief values (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORV to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be operable.

RELIEF VALVES (Continued)

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Operation with the block valves opened is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed to isolate PORV seat leakage is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The OPERABILITY of the PORVs and block valves in MODES 1, 2, and 3 is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,

2. Manual control of PORVs to control RCS pressure as required for SGTR mitigation,

3. Manual closing of a block valve to isolate a stuck open PORV,

4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and

5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for SGTR mitigation.

The non-safety PORV and block valve are used only as a backup to the two redundant safety grade PORVs and block valves to control RCS pressure for accident mitigation. Therefore, continued operation with the non-safety PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their safety functions. Surveillance Requirements 4.4.4.1 and 4.4.4.3 address the PORVs and Surveillance Requirement 4.4.4.2 addresses the block valves.

Surveillance Requirement 4.4.4.1.a provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for the PORV to be OPERABLE in MODES 1, 2, and 3.

Surveillance Requirement 4.4.4.1.b provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS. This test will be done in MODES 3 or 4, before the PORV is required for overpressure protection in TS 3.4.9.4.

RELIEF VALVES (Continued)

Surveillance Requirements 4.4.4.3 provides assurance of operability of the accumulators and that the accumulators are capable of supplying sufficient air to operate PORV(s) if they are needed for RCS pressure control and normal air and nitrogen systems are not available.

Surveillance Requirements 4.4.4.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with ACTION statements "b" or "c". This precludes the need to cycle the valves with a full system differential pressure or when maintenance is being performed to restore an inoperable PORV to OPERABLE status.

3/4.4.5 STEAM GENERATOR (SG)TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary-to-secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "Reactor Coolant Loops and Coolant Circulation, Startup and Power Operation," LCO 3.4.1.2, "Reactor Coolant System, Hot Standby," LCO 3.4.1.3, "Reactor Coolant System, Hot Shutdown," and LCO 3.4.1.4.1, "Reactor Coolant System, Cold Shutdown-Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.4.1, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.1, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operationalleakage. The SG performance criteria are described in Specification 6.8.4.1. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Reference 1).

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to 1 gallon per minute (gpm), plus the leakage rate associated with a double-ended rupture of a single tube. The accident radiological analysis for a SGTR assumes the ruptured SG secondary fluid is released directly to the atmosphere due to a failure of the PORV in the open position.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In some analyses developed by the industry, the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm, or is assumed to increase to 1 gpm as a result of accident induced conditions. The HNP accident analyses assume the amount of primary-to-secondary SG tube leakage is 1 gpm. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of 10 CFR 50.67 (Reference 2).

SG tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this specification, a SG tube is defined as the entire length of the tube, including the tube wall between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.l and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity preformance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients includued in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load verses displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collaspe. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions), and Service Level B (upset conditions), transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Reference 3) and Draft Regulatory Guide 1.121 (Reference 4).

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm total from all SGs. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2 and limits primary-to-secondary leakage through any one SG to 150 gpd. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

SG tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODES 1,2,3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1,2,3, and 4. In MODES 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

The ACTIONS are modified by a note clarifying that the Conditions may be entered independently for each SG tube. This clarification is acceptable because the required ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the required ACTIONS may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated required ACTIONS.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

ACTIONS a.1 and a.2

ACTIONS a.1 and a.2 apply if it is discovered that one or more SG tubes examined in an Inservice Inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. SG tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, ACTION b. applies.

An allowed completion time of seven days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, ACTION a.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This allowed completion time is acceptable since operation until the next inspection is supported by the operational assessment.

ACTION b.

If the requirements and associated completion time of ACTION a. are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. The allowed completion times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Surveillance Requirements

4.4.5.1 During shutdown periods, the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections, a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the method used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or area of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 5). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, specification 6.8.4.l contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by specification 6.8.4.l until subsequent inspections support extending the inspection interval.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

4.4.5.2 During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in specification 6.8.4.l are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of "Prior to entering HOT SHUTDOWN following a SG inspection" ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. 10 CFR 50.67
3. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
4. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976
5. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines"

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Reference 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Reference 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amout of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakages from these systems should be detected, located and isolated from containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant system pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. In some analyses developed by the industry, the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gallon per minute (gpm) or is assumed to increase to 1 gpm as a result of accident induced conditions. The HNP accident analyses assume the amount of primary-to-secondary SG tube leakage is 1 gpm. This 1 gpm leak rate includes the primary-to-secondary leakage rate existing immediately prior to the accident plus any additional increase in primary-to-secondary leakage induced during the accident. The LCO requirement to limit primary-to-secondary leakage through any one SG is limited to less than or equal to 150 gpd, which is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident or a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR analysis for a SGTR assumes the contaminated secondary fluid is released directly to the atmosphere due to a failure of the PORV in the open position and will continue atmospheric release until the time that the PORV can be isolated. The FSAR analysis for the SLB assumes that the SG with the failed steam line boils dry releasing all of the iodine directly to the environment and that iodine carried over to the faulted SG by tube leaks are also released directly to the environment until the RCS has cooled to below 212 degrees F. The dose consequences resulting from the SGTR and the SLB accidents are within the limits defined in 10 CFR 50.67.

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. UNIDENTIFIED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

c. PRIMARY-TO-SECONDARY LEAKAGE THROUGH ANY ONE STEAM GENERATOR

The limit of 150 gpd per SG is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 3). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tubeleakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of SG tube ruptures.

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or CONTROLLED LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

e. CONTROLLED LEAKAGE

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analysis.

f. REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAKAGE

The maximum allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping, which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Applicability

In MODES 1,2,3, and 4, the potential for RCPB leakage is greatest when the RCS is pressurized.

In Modes 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

ACTIONS

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

   The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

b. UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE in excess of the LCO limits must be reduced to within the limits within 4 hours. This completion time allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. Otherwise, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. This action is necessary to prevent further deterioration of the RCPB.

c. With RCS Pressure Isolation Valve leakage in excess of the limit, the high pressure portion of the affected system must be isolated within 4 hours, or be in at least HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. This action is necessary to prevent over pressurization of low pressure systems, and the potential for intersystem LOCA.

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Surveillance Requirements

4.4.6.2.1 Verifying RCS leakage to be within the LCO limits ensures that the integrity of the RCPB is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady-state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by a note. The note states that this SR is not required to be performed until 12 hours after establishing steady-state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady-state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and reactor cavity sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System Leakage Detection Systems."

Part (d) notes that this SR is not applicable to primary-to-secondary leakage. This is because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72-hour frequency is a reasonable interval to trend leakage and recognizes the importance of early detection in the prevention of accidents.

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

4.4.6.2.2 The Surveillance Requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

4.4.6.2.3 This SR verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one SG. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 4. The operational leakage rate limit applies to leakage through any one SG. If it is not practical to assign the leakage to an individual SG, all the primary-to-secondary leakage should be conservatively assumed to be from one SG.

The surveillance is modified by a note, which states that the Surveillance is not required to be performed until 12 hours after the establishment of steady-state operation. For RCS primary-to-secondary leakage determination, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance wih the EPRI guidelines (Reference 4).

References

1. 10 CFR 50, Appendix A, GDC 30

2. Regulatory Guide 1.45, May 1973

3. NEI 97-06, "Steam Generator Program Guidelines"

4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the SHEARON HARRIS site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1.0 microCurie/gram DOSE EQUIVALENT I-131, but less than 60.0 microCurie/gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. See Generic Letter 85-19 for additional information.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1.0 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 15 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 15 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, ASME Code Case N-640, and 10 CFR 50 Appendix G and H. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1).

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation: and

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 36 effective full power years (EFPY) of service life.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT}, including margin, computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

TABLE B 4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	GRADE	HEAT NO	Cu (wt.%)	Ni (wt.%)	T_{NDT} (°F)	INITIAL RT_{NDT} (°F)	CHARPY UPPER SHELF ENERGY TRANSVERSE FT-LB
Closure Hd. Dome	A533,B,CL1	A9213-1	-	-	-10	8	114
Head Flange	A508, CL2	5302-V2	-	-	0	0	135
Vessel Flange	"	5302-V1	-	-	-10	-8	110
Inlet Nozzle	"	438B-4	-	-	-20	-20	169
" "	"	438B-5	-	-	0	0	128
" "	"	438B-6	-	-	-20	-20	149
Outlet Nozzle	"	439B-4	-	-	-10	-10	151
" "	"	439B-5	-	-	-10	-10	152
" "	"	439B-6	-	-	-10	-10	150
Nozzle Shell	A533B,CL1	C0224-1	.12	-	-20	-1	90
" "	"	C0123-1	.12	-	0	42	84
Inter. Shell*	"	A9153-1	.09	.46	-10	60	83
" "*	"	B4197-2	.09	.50	-10	91	71
Lower Shell*	"	C9924-1	.08	.47	-10	54	98
" "*	"	C9924-2	.08	.47	-20	57	88
Bottom Hd. Torus	"	A9249-2	-	-	-40	14	94
" " Dome		A9213-2	-	-	-40	-8	125
Weld (Inter & Lower Shell Vertical Weld Seams)*		4P4784	.05	.91	-20	-20	>94
Weld (Inter. to Lower Shell Girth Seam)*		5P6771	.03	.94	-80	-20	80

* For Beltline Materials, copper and nickel valves are "best estimates".

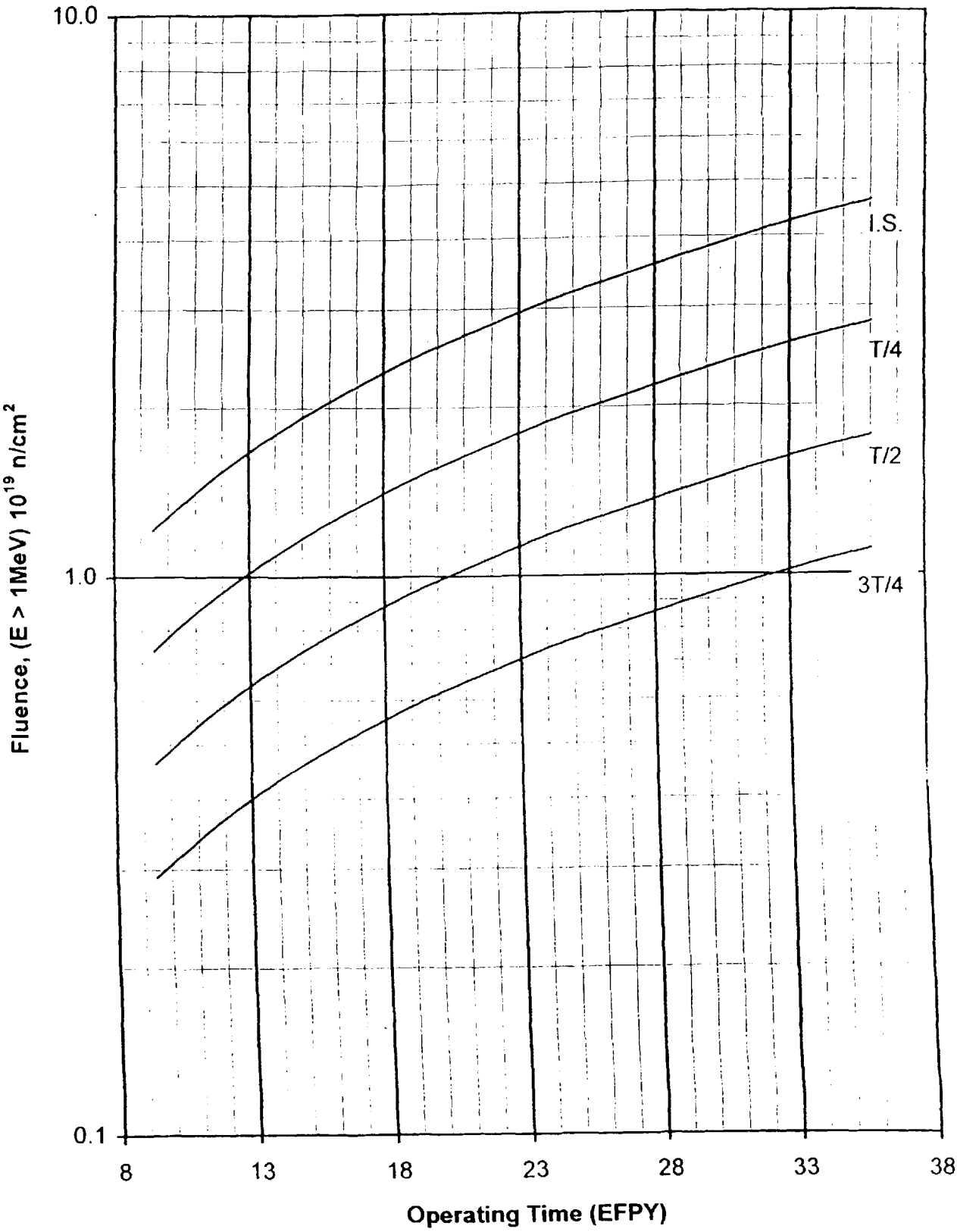

FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

FIGURE B 3/4.4-2 DELETED

PRESSURE/TEMPERATURE LIMITS (Continued)

The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 are based upon an adjusted RT_{NDT} (initial RT_{NDT} plus predicted adjustments for this shift in RT_{NDT} plus margin).

In accordance with Regulatory Guide 1.99, Revision 2, the results from the material surveillance program, evaluated according to ASTM E185, may be used to determine ΔRT_{NDT} when two or more sets of credible surveillance data are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and ASME Code Case N-640 for the reactor vessel controlling material.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures for the beltline shell region a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. A semielliptical inside corner flaw is assumed for the nozzle regions with a depth of one-quarter of the nozzle belt wall thickness. The inlet nozzle is used in the calculation procdures since the inner radius of this tapered nozzle is larger at the corner than the inner radius of the more tapered outlet nozzle. The dimensions of these postulated cracks, referred to in Appendix G of ASME Section XI as reference flaws, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

PRESSURE/TEMPERATURE LIMITS (Continued)

metal temperature at that time. K_{IR} is obtained from reference fracture toughness curves defined in the ASME Code. Pressure-temperature limits are developed for the vessel using the K_{IR} curve defined in Appendix A to the ASME Code, as permitted by ASME Code Case N-640. For the remaining components of the primary pressure boundary, pressure-temperature limits are based on the K_{IR} curve defined in Appendix G to the ASME Code. The K_{IR} curves are given by the equations:

Vessel regions:

$$K_{IR} = K_{Ic} = 33.2 + 2.806 \exp [0.02(T-RT_{NDT} + 100°F)] \quad (1a)$$

Remaining regions:

$$K_{IR} = K_{Ia} = 26.8 + 1.233 \exp [0.0145(T-RT_{NDT} + 160°F)] \quad (1b)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice leak and hydrostatic (ISLH) test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. The pressure stress intensity factors are obtained and allowable pressures are calculated from equation 2.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall and the inlet nozzle corner. During cooldown, the controlling location of the flaw is always at the inside surface because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest. The composite limit curves are developed considering the controlling reactor vessel component, either the beltline shell or the inlet nozzle.

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T inside surface location is at a higher temperature than the fluid adjacent to the inside surface. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside surface. The thermal gradients during heatup produce compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

The composite curves for the heatup rate data and the cooldown rate data in Figures 3.4-2 and 3.4-3 have not been adjusted for possible errors in the pressure and temperature sensing instruments. However, the heatup and cooldown curves in plant operating procedures have been adjusted for these instrument errors. The instrument errors are controlled by the Technical Specification Equipment List Program, Plant Procedure PLP-106.

"ISLH" pressure-temperature (P-T) curves may be used for inservice leak and hydrostatic tests with fuel in the reactor vessel. However, ISLH tests required by the ASME code must be completed before the core is critical.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.9 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 325°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of a charging/safety injection pump and its injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

of the LTOPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. LTOP instrument uncertainties are controlled by the Technical Specification Equipment List Program, Plant Procedure PLP-106. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one charging/safety injection pump while in MODES 4 (below 325°F), 5 and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and the reactor vessel service life.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition and Addenda through Summer 1978.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737,"Clarification of TMI Action Plant Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The value of 66% indicated level ensures that a minimum of 7440 gallons is maintained in the accumulators. The maximum indicated level of 96% ensures that an adequate volume exists for nitrogen pressurization.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed or boron concentration not within limits minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on the available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis demonstrates that the accumulators do not discharge following a large steam line break for HNP. Therefore, 72 hours is permitted to return the boron concentration to within limits. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 AND 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one charging/safety injection pump to be OPERABLE and the Surveillance Requirement to verify one charging/safety injection pump OPERABLE below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

ECCS SUBSYSTEMS (Continued)

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection into the core by the ECCS. This borated water is used as cooling water for the core in the event of a LOCA and provides sufficient negative reactivity to adequately counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertant depressurization, a LOCA, or a steam line rupture.

The limits on RWST minimum volume and boron concentration assure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all shutdown and control rods inserted except for the most reactive control assembly. These limits are consistent with the assumption of the LOCA and steam line break analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

An RWST allowed outage time of 12 hours is permitted during performance of Technical Specification surveillance 4.4.6.2.2 with a dedicated attendant stationed at valve 1CT-22 in communication with the Control Room. The dedicated attendant is to remain within the RWST compartment whenever valve 1CT-22 is open during the surveillance. The dedicated attendant can manually close valve 1CT-22 within 30 minutes in case of a line break caused by a seismic event. Due to the piping configuration, a break in the non-seismic portion of piping during this surveillance could result in draining the RWST below the minimum analyzed volume.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a, during performance of the periodic test, to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50, Option A for Type B and C tests, and the Containment Leakage Rate Testing Program for Type A tests.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

Action statement "a" has been modified by a note. The note allows use of the air lock for entry and exit for seven days under administrative controls if both air locks have an inoperable door. This seven day restriction begins when a door in the second air lock is discovered to be inoperable. Containment entry may be required to perform Technical Specification surveillances and actions, as well as other activities on equipment inside containment that are required by Technical Specifications (TS) or other activities that support TS required equipment. In addition, containment entry may be required to perform repairs on vital plant equipment, which if not repaired, could lead to a plant transient or a reactor trip. This note is not intended to preclude performing other activities (i.e., non-TS required activities or repairs on non-vital plant equipment) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize containment during the short time that an OPERABLE door is expected to be open.

CONTAINMENT AIR LOCKS (Continued)

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. HNP has an approved exemption to Appendix J Option A, paragraph III.D.2 of 10 CFR 50 in that the Overall air lock leakage test is required to be performed if maintenance has been performed that could affect the air lock sealing capability prior to establishing CONTAINMENT INTEGRITY. This is in contrast to the Appendix J requirement if air locks are opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of -2 psig, and (2) the containment peak pressure does not exceed the design pressure of 45 psig.

The maximum peak pressure expected to be obtained from a postulated LOCA is 41.8 psig using a value of 1.6 psig for initial positive containment pressure. The -1" wg was chosen to be consistent with the initial assumptions of the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of a postulated LOCA (41.8 psig). A visual inspection in conjunction with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment preentry purge makeup and exhaust isolation valves are required to be sealed closed during plant operations in MODES 1, 2, 3 and 4 since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during these MODES ensures that excessive quantities of radioactive materials will not be released via the Pre-entry Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Normal Containment Purge System is restricted to the 8-inch purge makeup and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during normal containment PURGING operation. The total time the Normal Containment Purge System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4.

Leakage integrity tests with a maximum allowable leakage rate for containment purge makeup and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before

CONTAINMENT VENTILATION SYSTEM (Continued)

gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Fan Coolers are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

The maximum and minimum volumes for the Spray Additive Tank are based on the analytical limits. The specified indicated levels used for surveillance include instrument uncertainties and unusable tank volume.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Fan Coolers ensures that adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

ESW flowrate to the Containment Fan Coolers will vary based on reservoir level. Acceptable ESW flowrate is dependent on the number of heat exchanger tubes in service. Surveillance test acceptance criteria should be adjusted for these factors.

The Containment Fan Coolers and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere.

As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Fan Coolers have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Reopening of an inoperable containment isolation valve is allowed to permit surveillance testing to demonstrate its operability of the operability of other equipment per Specification 4.6.3.1, or to change to compliance with another action statement for the LCO. An example of choosing an alternate action statement would be installing a blind flange versus using the failed closed containment isolation valve to isolate the penetration. This action would facilitate repair of the failed isolation valve, then removing the blind flange and re-installing the repaired valve. This process is acceptable because it results in restoring the penetration to its design configuration sooner that waiting for a plant shutdown to complete the repairs.

3/4.6.4 COMBUSTIBLE GAS CONTROL

Deleted.

3/4.6.5 VACUUM RELIEF SYSTEM

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than -1.93 psig. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of -2 psig.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.36×10^7 lbs/h which is in excess of 105% of the maximum calculated steam flow of 12.9×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 3 loop operation

$$HiØ = (100/Q)\frac{(w_s\ h_{fg}\ N)}{K}$$

Where:

HiØ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{(Btu/sec)}{Mwt}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec.

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

The values from this algorithm must then be adjusted lower to account for instrument and channel uncertainties. This adjustment will be 9% power.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions so that the Residual Heat Removal System may be placed into

AUXILIARY FEEDWATER SYSTEM

operation. The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15. The minimum pump performance requirements are based upon a maximum allowable degradation of the pump performance curves. Pump operation at this level has been demonstrated by calculation to deliver sufficient AFW flow to satisfy the accident analysis acceptance criteria.

With regard to the periodic AFW valve position verification of Surveillance Requirement 4.7.1.2.1 Sub-paragraph b.1, this requirement does not include in its scope the AFW flow control valves inline from the AFW motor-driven pump discharge header to each steam generator when they are equipped with an auto-open feature. The auto-open logic feature is designed to automatically open these valves upon receipt of an Engineered Safety Features System AFW start signal. As a consequence, valves with an auto-open feature do not have a "correct position" which must be verified. The valves may be in any position, in any MODE of operation thereby allowing full use of the AFW System for activities such as to adjust steam generator water levels prior to and during plant start-up, as an alternate feedwater system during hot standby, for cooldown operations, and to establish and maintain wet layup conditions in the steam generators.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 6 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics, and the value has also been adjusted in a manner similar to that for the RWST and BAT, as discussed on page B 3/4 1-3.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F (a generic maximum) and are sufficient to prevent brittle fracture. The Shearon Harris specific RT_{NDT} is limited to a maximum value of 10°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

The OPERABILITY of the Emergency Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," Rev. 2, January 1976.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

BACKGROUND

The CREFS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The CREFS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREFS train consists of a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provides backup in case of failure of the main HEPA filter bank.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and acident conditions. This area encompasses the control room, and may

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (continued)

BACKGROUND (continued)

encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILTIY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREFS is an emergency system, parts of which may also operate during normal unit operation in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demisters and heater are important to the effectiveness of the charcoal adsorbers.

Actuation of the CREFS places the system in the emergency mode (i.e., isolation with recirculation mode) of operation. Actuation of the system closes the unfiltered outside air intake and the unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and charcoal filters. The emergency mode also allows for pressurization and filtered ventilation of the air supply to the CRE.

Outside air is diluted with air from the CRE, filtered, and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. The air entering the CRE is continously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the emergency mode.

A single CREFS train operating with a maximum pressurization flow rate of 400 cfm will pressurize the CRE to at least 0.125 inches water gauge relative to external areas adjacent to the CRE boundary. The CREFS operation in maintaining the CRE habitable is discussed in the FSAR, Section 9.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category 1 requirements.

The CREFS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem TEDE or its equivalent to any part of the body.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (continued)

APPLICABLE SAFETY ANALYSIS

The CREFS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREFS provides airborne radiological protection for the occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident fission product release presented in the FSAR, Chapter 15 (Ref. 2).

The CREFS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of toxic chemical hazards found no impact on control room habitability from toxic chemical sources (Ref.3). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 4).

The worst case single active failure of a component of the CREFS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREFS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITION FOR OPERATION (LCO)

Two independent and redundant CREFS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE or its equivalent to any part of the body to the CRE occupants in the event of a large radioactive release.

Each CREFS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREFS train is OPERABLE when the associated:

a. Fan is OPERABLE.
b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions, and
c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREFS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

3/4.7.6 Control Room Emergency Filtration System (continued)

LIMITING CONDITION FOR OPERATION (LCO)

A failure to secure the RAB Normal Ventilation System, as part of a control room isolation, results in an inoperable control room boundary. Various postulated alignments or malfunctions of the RAB Normal Ventilation System can result in either excessively positive or negative RAB pressures, which can compromise the ability of the CREFS trains to maintain the control room envelope at a positive pressure of 1/8 INWG or greater relative to adjacent areas, thus directly impacting design basis in-leakage assumptions and personnel dose consequences under accident conditions.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, during movement of irradiated fuel assemblies, and during movement of loads over spent fuel pools, the CREFS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

During movement of irradiated fuel assemblies and movement of loads over spent fuel pools, the CREFS must be OPERABLE to cope with the release from a fuel handling accident involving irradiated fuel.

ACTIONS

3.7.6.a.1

In MODE 1, 2, 3, or 4, when one CREFS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this condition, the remaining OPERABLE CREFS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREFS train could result in loss of CREFS function. The 7 day allowed outage time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

In MODE 1, 2, 3, or 4, if the inoperable CREFS train cannot be restored to OPERABLE status within the allowed outage times, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least a MODE 3 within the next 6 hours, and in MODE 5 within the following 30 hours. The allowed outage times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

3.7.6.a.2

In MODE 1, 2, 3, or 4, if the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE, this can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE or its equivalent to any part of the body), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke.
These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional.

3.7.6.a.2 (continued)

The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day allowed outage time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day allowed outage time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

In MODE 1, 2, 3, or 4, if the inoperable CRE boundary cannot be restored to OPERABLE status within the allowed outage times, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within the next 6 hours, and in MODE 5 within the following 30 hours. The allowed outage times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

3.7.6 b.1 and c.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, or during movement of loads over spent fuel pools, when one CREFS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. If the inoperable CREFS train cannot be restored to OPERABLE status within the allowed outage time, action must be taken to immediately place the OPERABLE CREFS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

3.7.6 b.2 and c.2

When both CREFS trains are inoperable, for reasons other than an inoperable CRE boundary, or when the OPERABLE CREFS train required to be in the emergency mode by ACTION b.1 or c.1 is not capable of being powered by an OPERABLE emergency power source, immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

3.7.6 b.3 and c.3

In MODE 5 or 6, or during movement of irradiated fuel assemblies, or duringmovement of loads over spent fuel pools, with one or more CREFS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minmizes the accident risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 4.7.6.a

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

SR 4.7.6 b.c.e. and f

ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon a removal efficiency of 99% for elemental, particulate and organic forms of radioiodine.

SR 4.7.6 d.1

This SR verifies that the HEPA filters and chaarcoal adsorbers are not excessively blocked. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for those components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit. The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 4.7.6 d.2

This SR verifies that each CREFS train starts and operated on an actual or simulated actuation signal. The Frequency of 18 months is based in industry operating experience and is consistent with the typical refueling cycle.

SR 4.7.6 d.4

This SR verifies that each CREFS train heater operates within assumed parameters. The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 4.7.6.g

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the locensing basis analyses of DBA consequences is no more than 5 rem TEDE or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences.

In MODES 1, 2, 3, or 4, when unfiltered air inleakage is greater than the assumed flow rate, ACTION a.2 must be entered. ACTION a.2 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 5) which endorses, with exceptions NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by ACTION a.2. Temporary analytical methods may also be used a compensatory measures to restore OPERABILTIY (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

1. FSAR, Section 9.4
2. FSAR, Chapter 15
3. FSAR, Section 6.4
4. FSAR, Section 9.5 and Corrective Action Program Assignment 100903-05.
5. Regulatory Guide 1.196
6. NEI 99-03, "Control Room Habitability Assessment," June 2001.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (Continued)

3/4.7.7 REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEM

The OPERABILITY of the Reactor Auxiliary Building Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

The LCO is modified by a note allowing the Reactor Auxiliary Building Emergency Exhaust System (RABEES) ventilation boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for RABEES isolation is indicated.

If the RABEES boundary is inoperable in MODES 1, 2, 3, and 4, the RABEES trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE RABEES boundary within 24 hours. During the period that the RABEES boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19, 60, 64, and 10 CFR Part 100) should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns. HNP will have written procedures available describing compensatory measures to be taken in the event of an intentional or unintentional entry into this condition. The 24 hour allowed out of service time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the RABEES boundary. The 24 hour allowed out of service time is reasonable based on the low probability of a DBA occurring during this time period, and the availability of compensatory measures.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Manager-Technical Support. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

Surveillance to demonstrate OPERABILITY is by performance of an augmented inservice inspection program.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.9 SEALED SOURCE CONTAMINATION

The sources requiring leak tests are specified in 10 CFR 31.5(c)(2)(ii). The limitation on removable contamination is required by 10 CFR 31.5(c)5. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources that are frequently handled are required to be tested more often than those that are not. Sealed sources that are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 DELETED

3/4.7.11 DELETED

3/4.7.12 DELETED

3.4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

The OPERABILITY of the Emergency Service Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.14 FUEL STORAGE POOL BORON CONCENTRATION

The fuel storage pools contain several rack designs. The PWR and BWR racks in Pools "C" and "D" have a poison that maintains k_{eff} less than or equal to 0.95 during normal operation. The BWR racks in Pools "A" and "B" also credit a poison in the rack design. For the PWR racks in Pools "A" and "B", the installed poison is not credited and soluble boron is relied upon to maintain the storage K_{eff} less than or equal to 0.95 during normal operation. Soluble boron is also relied upon during design basis accidents (e.g. fuel handling accident (FHA) or misloading) to maintain k_{eff} less than or equal to 0.95. The most limiting boron requirement is 1000 ppm of any of the pools. The difference between 2000 ppm and 1000 ppm provides margin for boron measurement uncertainties and the detection and mitigation of an accidental boron dilution event. It is not required to postulate the boron dilution accidents concurrent with another accident such as fuel misloading or FHA.

The water in the pools normally contains a concentration in excess of 2000 ppm. The pools are typically interconnected through canals. Years of operating data show that the boron concentration does not vary significantly from pool to pool. The sampling surveillance permits taking a sample from any location in the connected volume of the pools. This is typically done by rotating between four widely separated locations (e.g. Pool A, Pool B, Pool C and 1&4 Transfer Canal) in the entire pool volume. Sampling of an individual pool is only required when a specific pool is isolated such that diffusion of the boron between pools is restricted.

3/4.8.1, 3/4.8.2, AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The switchyard is designed using a breaker-and-a-half scheme. The switchyard currently has seven connections with the Duke Energy transmission network; each of these transmission lines is physically independent. The switchyard has one connection with each of the two Startup Auxiliary Transformers and each SAT can be fed directly from an associated offsite transmission line. The Startup Auxiliary Transformers are the preferred power source for the Class 1E ESF buses. The minimum alignment of offsite power sources will be maintained such that at least two physically independent offsite circuits are available. The two physically independent circuits may consist of any two of the incoming transmission lines to the SATs (either through the switchyard or directly) and into the Class 1E system. As long as there are at least two transmission lines in service and two circuits through the SATs to the Class 1E buses, the LCO is met.

During MODES 5 and 6, the Class 1E buses can be energized from the offsite transmission network via a combination of the main transformers and unit auxiliary transformers. This arrangement may be used to satisfy the requirement of one physically independent circuit.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. There are additional ACTION requirements to verify that all required feature(s) that depend on the remaining OPERABLE A.C. sources as a source of emergency power, are also OPERABLE. These requirements allow a period of time to restore any required feature discovered to be inoperable, e.g. out-of-service for maintenance, to an OPERABLE status. If the required feature(s) cannot be restored to an OPERABLE status, the ACTION statement requires the redundant required feature, i.e. feature receiving power from an inoperable A.C. source, to be declared inoperable. The allowed operating times to restore an inoperable required feature to an OPERABLE status is based on the requirements in NUREG 1431. The term "verify", as used in these ACTION statements means to administratively check by examining logs or other information to determine the OPERABILITY of required feature(s). It does not mean to perform the Surveillance Requirement needed to demonstrate the OPERABILITY of the required feature(s).

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based upon the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; 1.108, "Periodic Testing of Diesel

Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 as modified in accordance with the guidance of IE Notice 85-32, April 22, 1985; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Proper shedding and sequencing of loads are required functions for Emergency Diesel Generator OPERABILITY. Pressure testing of the diesel generator fuel oil piping at 110% of the system design pressure will only be required on the isolable portions of (1) fuel oil transfer pump discharge piping to the day tank, (2) fuel oil supply from the day tank to the diesel vendor-supplied piping, and (3) fuel oil return piping from the diesel vendor-supplied piping to the day tank regulator valve. The exemptions allowed by ASME Code Section XI will be invoked for the atmospheric day tanks and non-isolable piping.

The inclusion of the loss of generator potential transformer circuit lockout trip is a design feature based upon coincident logic and is an anticipatory trip prior to diesel generator overspeed. In TS 4.8.1.1.2.f.13, the phrase “all diesel generator trips” refers to automatic protective trips.

The Surveillance Requirements for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." The performance test supporting the Surveillance Requirement incorporates the guidance of IEEE Std 450-2010.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

LCOs 3.8.3.1 and 3.8.3.2 include requirements for energizing 118 VAC vital buses from the associated inverters connected to 125 VDC buses. In the event the 118 VAC vital buses are not energized by the inverters connected to the 125 VDC buses, system design provides for

energizing the 118 VAC buses from the Bypass Source or the Alternate Power Supply. The Bypass Source is regulated, transfer to the source is automatic within the inverters, and operation on the Bypass Source requires entry into LCO 3.8.3.1 Action ‘c’ or LCO 3.8.3.2 Action, depending on the OPERATIONAL MODE. The Alternate Power Supply is unregulated and the voltage may not be sufficient to support loads as documented in calculation E-6007. Operation on the Alternate Power Supply, requires entry into LCO 3.8.3.1 Action ‘b’ or LCO 3.8.3.2 Action, depending on the OPERATIONAL MODE.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes. For surveillance 4.8.4.1.a.1.c and 4.8.4.1.a.2, testing of the breakers includes a representative sample of 10% of each type of breaker as described in the table below.

Types
15-Amp(A)
30A-40A
50A
60A
70A-90A
100-110A
125-150A
225A

The bypassing of the motor-operated valves thermal overload protection during accident conditions by integral bypass devices ensures that safety-related valves will not be prevented from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection during accident conditions are in accordance with Regulatory Guide 1.106, “Thermal Overload Protection for Electric Motors on Motor Operated Valves,” Revision 1, March 1977.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses and are specified in the cycle-specific COLR. The boron concentration limit specified in the COLR ensures that a core K_{eff} of $\leq$ 0.95 is maintained during fuel handling operations. The administrative controls over the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors and/or Wide Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. If the audible indication is lost, then enter LCO Action 3.9.2.b.

3/4.9.3 DECAY TIME - DELETED

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Penetrations applicable to Technical Specification 3.9.4.b and 3.9.4.c may be opened provided the following administrative controls are in effect:

1. An individual or individuals shall be designated and available at all times, capable of isolating the breached penetration.

2. The breached penetrations shall not be obstructed unless capability for rapid removal of obstructions is provided (such as quick disconnects for hoses).

3. For the Personnel Air Lock, at least one door must be capable of being closed and secured. Additionally, the equipment hatch must be capable of being closed and secured. Equivalent isolation methods may also be used.

The LCO is modified by a Note allowing penetration flow paths providing direct access from the containment atmosphere to the outside atmosphere to be open under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

CONTAINMENT BUILDING PENETRATIONS (Continued)

The allowance to have containment penetration (including the airlock doors and equipment hatch) flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on (1) confirmatory dose calculations as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident that the airlock or equipment hatch can and will be promptly closed following containment evacuation (even though the containment fission product control function is not required to meet acceptable dose consequences) and that the open penetration(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

Containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated, or capable of isolation via administrative controls, on at least one side of containment. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movement.

3/4.9.5 COMMUNICATIONS - DELETED

3/4.9.6 REFUELING MACHINE - DELETED

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING - DELETED

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

The minimum RHR flow requirement is reduced to 900 gpm when the reactor water level is below the reactor vessel flange. The 900 gpm limit reduces the possibility of cavitation during operation of the RHR pumps and ensures sufficient mixing in the event of a MODE 6 boron dilution incident.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge makeup and exhaust penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

CONTAINMENT VENTILATION ISOLATION SYSTEM (Continued)

Penetrations applicable to Technical Specification 3.9.9 may be opened provided the following administrative controls are in effect:

1. An individual or individuals shall be designated and available at all times, capable of isolating the breached penetration.

2. The breached penetration shall not be obstructed unless capability for rapid removal of obstructions is provided (such as quick disconnects for hoses).

The LCO is modified by a Note allowing penetration flow paths providing direct access from the containment atmosphere to the outside atmosphere to be open under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

The allowance to have containment penetration (including the airlock doors and equipment hatch) flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on 1) confirmatory dose calculations as approved by the NRC staff which indicate acceptable radiological consequences and 2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident that the airlock or equipment hatch can and will be promptly closed following containment evacuation (even though the containment fission product control function is not required to meet acceptable dose consequences) and the open penetration(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND NEW AND SPENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

According to Regulatory Guide 1.25, Revision 0, there is 23 feet of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 feet of water, the assumptions of Regulatory Guide 1.25, Revision 0, can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontal on top of the spent fuel racks; however, there may be <23 feet of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail.

3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

The limitations on the Fuel Handling Building Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

The LCO is modified by a note allowing the Fuel Handling Building Emergency Exhaust System (FHBEES) ventilation boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for FHBEES isolation is indicated.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of shutdown and control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual shutdown and control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure shutdown and control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DELETED

3/4.11.2.2 DELETED

3/4.11.2.3 DELETED

3/4.11.2.4 DELETED

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GASEOUS RADWASTE TREATMENT SYSTEM downstream of the hydrogen recombiners is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of oxygen to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 DELETED

3/4.11.3 DELETED

3/4.11.4 DELETED

Pages B 3/4 11-3 through B 3/4 11-6 have been deleted.

3/4.12.1 DELETED

3/4.12.2 DELETED

3/4.12.3 DELETED

Page B 3/4 12-2 has been deleted.