Turkey Point Units 3 and 4

LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE

ATTACHMENT 3

Technical Specifications Bases Markups
for Information Only

This coversheet plus 35 pages

ATTACHMENT 1
(Page 3 of 112)

TECHNICAL SPECIFICATION BASES

2.1.1 (Cont'd)

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, and a reference cosine with a peak of ~~1.55~~ 1.78 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H}[1+ PF_{\Delta H} (1-P)]$$

Where P is the fraction of RATED THERMAL POWER

$F^{RTP}_{\Delta H}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT.

$PF_{\Delta H}$ = Power Factor multiplier for $F_{\Delta H}$ as specified in the CORE OPERATING LIMITS REPORT.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit assuming the axial power imbalance is within the limits of the f (ΔT) function of the Overtemperature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.

Fuel rod bowing reduces the values of DNB ratio (DNBR). The penalties are calculated pursuant to Fuel Rod Bow Evaluation, WCAP-8691-P-A Revision 1 (Proprietary) and WCAP-8692 Revision 1 (Non-Proprietary). The restrictions of the Core Thermal Hydraulic Safety Limits assure that an amount of DNBR margin greater than or equal to the above penalties is retained to offset the rod bow DNBR penalty.

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 vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The RCS piping, valves and fittings are designed to ANSI B31.1, which permits a maximum transient pressure of 120% of design pressure of 2485 psig. The Safety Limit of 2735 psig is therefore more conservative than the ANSI B31.1 design criteria and consistent with associated ASME Code requirements.

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

ATTACHMENT 1
(Page 4 of 112)

TECHNICAL SPECIFICATION BASES

2.2 Limiting Safety System Settings

2.2.1 Reactor Trip System Instrumentation Setpoints

The Trip Setpoints or Nominal Trip Setpoints (NTS)

~~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 Trip Setpoint when the as measured setpoint is within the band allowed for calibration accuracy.

17070-P (for functional units 2a, 5, 6, 10, 11, and 12)

To accommodate ~~the~~ instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, ~~statistical~~ allowances are provided for in the Nominal Trip Setpoint and Allowable Values in accordance with the setpoint methodology described in WCAPs ~~12201~~ and 12745. Surveillance criteria have been determined and are controlled in Plant procedures and in design documents. The surveillance criteria ensure that instruments which are not operating within the assumptions of the setpoint calculations are identified. An instrument channel is considered OPERABLE when the surveillance is within the Allowable Value and the channel is capable of being calibrated in accordance with Plant ~~procedures.~~ Sensor and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

17070-P

The inability to demonstrate through measurement and/or analytical means, using the methods described in WCAPs ~~12201~~ and 12745 ~~(TA>R+S+Z)~~, that the Reactor Trip function would have occurred within the values specified in the design documentation provides a threshold value for REPORTABLE EVENTS.

There is a small statistical probability that a properly functioning device will drift beyond determined surveillance criteria. Infrequent drift outside the surveillance criteria are expected. Excessive rack or sensor drift that is more than occasional may be indicative of more serious problems and should warrant further investigations.

INSERT B2.2.1 INSERT HERE

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.

a

procedures to within the as-left tolerance. If the as-found setpoint is outside of the Plant Procedure as-found tolerance, the occurrence will be entered into the plant Corrective Action Program and the channel will be evaluated to verify that it is functioning as required before returning the channel to service.

INSERT FOR B2.2.1

The trip setpoints used in the bistables for Functional Units 2a, 5, 6, 10, 11, and 12 are based on the analytical limits stated in WCAP-17070-P. The selection of these Nominal Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49, the NTS are conservative with respect to the analytical limits. A detailed description of the methodology used to determine the NTS, Allowable Value, as-left tolerance, and as found tolerance including their explicit uncertainties, is provided in WCAP-17070-P which incorporates all of the known uncertainties applicable to these functional units. The magnitudes of these uncertainties are factored into the determination of each NTS and corresponding Allowable Value.

The NTS for Functional Units 2a, 5, 6, 10, 11, and 12 is the value at which the bistable is set and is the expected value to be achieved during calibration. The NTS value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable is considered to be properly adjusted when the as-left NTS value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e., ± rack calibration accuracy). The NTS value is therefore considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of COT and CHANNEL CALIBRATION. These functional Units have been modified by two notes as identified in Table 2.2-1. Note (a) requires evaluation of channel performance for the condition where the as found setting for the nominal trip setpoint is outside of the as found criterion. As stated above, these instances will be entered into the plant corrective action process and the channel will be evaluated to verify that it is functioning as required before returning the channel to service. Note (b) requires that the channel as left setting must be within the as left tolerance band. As noted above a channel is considered to be properly calibrated with the as left NTS value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e., ± rack calibration accuracy).

ATTACHMENT 1
(Page 5 of 112)

TECHNICAL SPECIFICATION BASES

2.2.1 (Cont'd)

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 for all power levels to mitigate the consequences of a reactivity excursion which may be too rapid for the temperature and pressure protective trips.

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.

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 ~~105~~ $\boxed{10^5}$ 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. No credit is taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

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 piping transit delays from the core to the temperature detectors 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 piping delays from the core to 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.

ATTACHMENT 1
(Page 7 of 112)

TECHNICAL SPECIFICATION BASES

2.2.1 (Cont'd)

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine ~~first stage~~ 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% of loop design flow. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of ~~nominal full~~ design loop flow. Conversely, on decreasing power between P–8 and the 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 Water Level-Low 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 ~~greater than or equal to 0.665 x 106 lbs/hour~~ 20% rated steam flow. The Steam Generator Water Level-Low portion of the trip is activated when the water level drops below ~~10%~~ 16%, 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 - 4.16 kV Bus A and B Trips

The 4.16 kV Bus A and B Undervoltage trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified setpoint assures a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. The delay is set so that the time required for a signal to reach the Reactor trip breakers following the trip of at least one undervoltage relay in both of the associated Units 4.16 kV busses shall not exceed ~~1.3~~ 1.5 seconds. On decreasing power the Undervoltage Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine ~~first stage~~ inlet pressure at approximately 10% of full power equivalent) and on increasing power, reinstated automatically by P-7.

ATTACHMENT 1
(Page 8 of 112)

TECHNICAL SPECIFICATION BASES

2.2.1 (Cont'd)

Turbine Trip

inlet

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

Insert "TS Basis page 19a"

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 Coolant Pump Breaker Position Trip

inlet

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine ~~first stage~~ pressure at approximately 10% of full power equivalent) an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7, an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Underfrequency sensors are also installed on the 4.16 kV busses to detect underfrequency and initiate breaker trip on underfrequency. The underfrequency trip setpoints preserve the coast down energy of the reactor coolant pumps, in case of a grid frequency decrease so DNB does not occur.

The Reactor trip from Turbine trip function anticipates the loss of secondary heat removal capability from a power level above the P-7 setpoint. Below the P-7 setpoint this action will not actuate a reactor trip. The Turbine Trip Function is not required to be OPERABLE below P-7 because load rejection can be accommodated by the steam dump system or Steam Generator Atmospheric Dump Valves. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore, there is no potential for a turbine trip.

Tripping the reactor in anticipation of a loss of secondary heat removal capability acts to minimize the pressure and temperature transients on the reactor. Two separate mechanisms are design to detect a turbine trip, low EHC oil pressure or Stop Valve closure. Three pressure switches monitor the control oil pressure in the Emergency Trip Header Oil in the EHC System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The Turbine Trip-Turbine Stop Valve Closure trip function is diverse to the Turbine Emergency Trip Header Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If both limit switches indicate that the stop valves are closed, a reactor trip is initiated. This Function only measures the discrete position (open or closed) of the turbine stop valves. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves. The reactor trip from Turbine trip is not credited in the accident analysis for core protection.

The setpoint is considered to be adjusted consistent with the Nominal Trip Setpoint (NTS) when the as-measured setpoint is within the band allowed for calibration accuracy. Notes (a) and (b) are applied to the Turbine Emergency Trip Header Pressure trip function in Table 2.2-1, Reactor Trip Setpoints and in Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements. Note (a) requires that if the as-found trip setpoint is outside predefined limits based on actual expected errors between calibrations then corrective action is required. If the as-found trip setpoints are outside the Allowable Value, then the instrument channel is inoperable. Note (a) requires that the trip setpoint is considered to be properly adjusted when the as-left NTS value is within the predefined as-left tolerance for CHANNEL CALIBRATION. The as-found and as-left values shall be determined using a methodology consistent with WCAP-17070

ATTACHMENT 1
(Page 20 of 112)

TECHNICAL SPECIFICATION BASES

3/4.1 Reactivity Control Systems

3/4.1.1 Boration Control

3/4.1.1.1 &
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}. 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. Figure 3.1-1 shows the SHUTDOWN MARGIN equivalent to 1.77% Δk/k at the end-of-core-life with respect to an uncontrolled cooldown. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With Tavg less than 200°F, the reactivity transients resulting from an inadvertent cooldown of the RCS or an inadvertent dilution of RCS boron are minimal and a 1% Δk/k SHUTDOWN MARGIN provides adequate protection.

The boron rate requirement of 16 gpm of 3.0 wt% (5245 ppm) boron or equivalent ensures the capability to restore the shutdown margin with one OPERABLE charging pump.

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; 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.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value ~~-3.5~~ x 10-4 Δk/k/°F. The MTC value of ~~-3.0~~ x 10-4 Δk/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of ~~-3.5~~ x 10-4 Δk/k/°F.

-3.5 surveillance value -4.1 -4.1

ATTACHMENT 1
(Page 22 of 112)

TECHNICAL SPECIFICATION BASES

3/4.1.2 (Cont'd)

The ACTION statement restrictions for the boration flow paths allow continued operation in mode 1 for a limited time period with either boration source flow path or the normal flow path to the RCS (via the regenerative heat exchanger) inoperable. In this case, the plant capability to borate and charge into the RCS is limited and the potential operational impact of this limitation on mode 1 operation must be addressed. With both the flow path from the boric acid tanks and the regenerative heat exchanger flow path inoperable, immediate initiation of action to go to COLD SHUTDOWN is required but no time is specified for the mode reduction due to the reduced plant capability with these flow paths inoperable.

Two charging pumps are required to be OPERABLE to ensure single functional capability in the event an assumed failure renders one of the pumps or power supplies inoperable. Each bus supplying the pumps can be fed from either the Emergency Diesel Generator or the offsite grid through a startup transformer.

The boration capability of either flow path is sufficient to provide the required SHUTDOWN MARGIN in accordance with Figure 3.1-1 from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL peak xenon conditions without letdown such that boration occurs only during the makeup provided for coolant contraction. This requirement can be met for a range of boric acid concentrations in the boric acid tank and the refueling water storage tank. The range of boric acid tanks requirements is defined by Technical Specification 3.1.2.5.

With the RCS temperature below 200°F, one boron injection source flow path is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection system source flow path becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,900 gallons of at least 3.0 wt% (5245 ppm) borated water per unit from the boric acid storage tanks or 20,000 gallons of ~~1950~~ ppm borated water from the RWST.

between 2400 and 2600

The charging pumps are demonstrated to be OPERABLE by testing as required by the ASME OM code or by specific surveillance requirements in the specification. These requirements are adequate to determine OPERABILITY because no safety analysis assumption relating to the charging pump performance is more restrictive than these acceptance criteria for the pumps.

ATTACHMENT 1
(Page 23 of 112)

TECHNICAL SPECIFICATION BASES

3/4.1.2 (Cont'd)

The boron concentration of the RWST in conjunction with manual addition of borax ensures that the solution recirculated within containment after a LOCA will be basic. The basic solution minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The temperature requirements for the RWST are based on the containment integrity and large break LOCA analysis assumptions.

The OPERABILITY of one Boron Injection flowpath during REFUELING ensures that this system is available for reactivity control while in MODE 6. Components within the flowpath, e.g., boric acid transfer pumps or charging pumps, must be capable of being powered by an OPERABLE emergency power source, even if the equipment is not required to operate.

The OPERABILITY requirement of ~~55~~ 62°F and corresponding surveillance intervals associated with the boric acid tank system ensures that the solubility of the boron solution will be maintained. The temperature limit of ~~55~~ 62°F includes a 5°F margin over the ~~50~~ 57°F solubility limit of ~~3.5~~ 4.0 wt.% boric acid. Portable instrumentation may be used to measure the temperature of the rooms containing boric acid sources and flow paths.

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. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the demand counter position. For the Shutdown Banks and Control Banks A and B, the Position Indication requirement is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 steps withdrawn and All Rods Out (ARO) inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these bank positions in these two areas satisfies all accident analysis assumptions concerning their position. For Control Banks C and D, the Position Indication requirement is defined as the group demand counter indicated position between 0 steps withdrawn and All Rods Out (ARO) inclusive.

ATTACHMENT 1
(Page 29 of 112)

TECHNICAL SPECIFICATION BASES

3/4.2.2 and 3/4.2.3 (Cont'd)

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. These uncertainties only apply if the map is taken for purposes other than the determination of P_{BL} and P_{RB}.

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

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \le F^{RTP}_{\Delta H}/1.08$, where $F^{RTP}_{\Delta H}$ is the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT. The logic behind the larger uncertainty in this case is that (a) Normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) Although the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) An error in the prediction for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, ~~bu~~ but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$, is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

The following are independent augmented surveillance methods used to ensure peaking factors are acceptable for continued operation above Threshold Power, P_T:

Base Load - This method uses the following equation to determine peaking factors:

$$F_{QBL} = F_Q(Z) \text{ measured} \times 1.09 \times W(Z)_{BL}$$

where: $W(Z)_{BL}$ = accounts for power shapes;
1.09 = accounts for uncertainty;
$F_Q(Z)$ = measured data;
F_{QBL} = Base load peaking factor.

ATTACHMENT 1
(Page 33 of 112)

TECHNICAL SPECIFICATION BASES

3/4.2.4 <u>Quadrant Power Tilt Ratio</u>

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 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 $F_Q(Z)$ 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 or incore thermocouple map 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 two 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.

3/4.2.5 <u>DNB Parameters</u>

2182

2204

270,000

585

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 above the applicable design limits throughout each analyzed transient. The indicated Tavg value of ~~581.2~~°F and the indicated pressurizer pressure value of ~~2200~~ psig correspond to analytical limits of ~~583.2~~°F and ~~2175~~ psig respectively, with allowance for measurement uncertainty.

589

The measured RCS flow value of ~~264,000~~ gpm corresponds to ~~an analytical limit~~ of ~~255,000~~ gpm ~~which is assumed to have a 3.5%~~ calorimetric measurement uncertainty.

260,700

with an allowance of
3.5% to accomodate

a Thermal Design Flow

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.2.5 (Cont'd)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to ensure that the DNB-related flow assumption is met and to ensure correlation of the flow indication channels with measured flow. ~~Six month drift effects have been included for feedwater temperature, feedwater flow, steam pressure, and the pressurizer pressure inputs. The flow measurement is performed within ninety days of completing the cross calibration of the hot leg and cold leg narrow range RTDs.~~ The indicated percent flow surveillance on a 12-hour basis will provide sufficient verification that flow degradation has not occurred. An indicated percent flow which is greater than the thermal design flow plus instrument channel inaccuracies and parallax errors is acceptable for the 12 hour surveillance on RCS flow. To minimize measurement uncertainties it is assumed that the RCS flow channel outputs are averaged.

3/4.3 Instrumentation

3/4.3.1 &
3/4.3.2 Reactor Trip System 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 is maintained, (3) Sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance (due to plant specific design, pulling fuses and using jumpers may be used to place channels in trip), and (4) 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. Surveillances for the analog RPS/ESFAS Protection and Control rack instrumentation have been extended to quarterly 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 generically approved by the NRC and documented in their SERs (Letters to the Westinghouse Owner's Group from the NRC dated February 21, 1985, February 22, 1989, and April 30, 1990).

Under some pressure and temperature conditions, certain surveillances for Safety Injection cannot be performed because of the system design. Allowance to change modes is provided under these conditions as long as the surveillances are completed within specified time requirements.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.3.1 and 3/4.3.2 (Cont'd)

If the reactor trip breakers (RTB) are closed and the Rod Control System is capable of withdrawing the control rods, then source range instrumentation is required to support Technical Specification 3.3.1, Table 3.3-1, Item 4c. This is specified by the single asterisk note and the requirement in the table for the trip function. Otherwise, Item 4b of Table 3.3-1 applies. The double asterisk note of Item 4b allows the use of the Gammametrics only if the RTBs are open. If the RTBs are closed but the Rod Control System is not capable of withdrawing rods, then Item 4b does not allow Gammametrics to take the place of source range instruments. Item 4b does not require the trip function to be operable.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-3 are the nominal values at which the bistables are set for each functional unit. The setpoint is considered to be adjusted consistent with the Nominal Trip Setpoint when the as measured setpoint is within the band allowed for calibration accuracy. Although the degraded voltage channel for Item 7.c consists of definite time (ITE) and inverse time (IAV) relays, the setpoint specified in Table 3.3-3 is only applicable to the definite time delay relays (Reference: CR 00-2301). The original protection scheme consisted of inverse time voltage relays; but based on operational experience, it was found that the settings of these relays drifted in a non-conservative direction. In 1992, to improve repeatability and to reduce potential harmful effects due to setpoint drifts, ITE definite time delay relays were added to the protection scheme to protect the 480 V alternating current (AC) system from adverse effects of a sustained degraded voltage condition. The IAV relays protect the system from adverse effects of a brief large voltage transient. The IAV relay settings are such that they should not operate before the ITE relays. The degraded voltage protection is ensured by the definite time delay relays with the setpoints specified in the TS Table 3.3-3, Item 7.c (References: L-92-097 dated 4/21/92, and L-92-215 dated 7/29/92). These changes were approved by NRC letter dated August 20, 1992, and implemented by Amendment Nos 152 and 147.

To accommodate ~~the~~ instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, ~~statistical~~ allowances are provided for in the Nominal Trip Setpoint and Allowable Values in accordance with the setpoint methodology described in WCAPs 17070-P (for functional units 1f, 4d, 5c, and 6b) ~~12201~~ and 12745. Surveillance criteria have been determined and are controlled in Plant procedures and in design documents. The surveillance criteria ensure that instruments which are not operating within the assumptions of the setpoint calculations are identified. An instrument channel is considered OPERABLE when the surveillance is within the Allowable Value and the channel is capable of being calibrated in accordance with Plant ~~procedures.~~ procedures to within the as-left tolerance. If the as-found setpoint is outside of the Plant Procedure as-found tolerance, the occurrence will be entered into the plant Corrective Action Program and the channel will be evaluated to verify that it is functioning as required before returning the channel to service. Sensor and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

The inability to demonstrate through measurement and/or analytical means, using the methods described in WCAPs 17070-P ~~12201~~ and 12745 ~~(TA ≥R+S+Z)~~, that the Reactor Trip function would have occurred within the values specified in the design documentation provides a threshold value for REPORTABLE EVENTS.

There is a small statistical probability that a properly functioning device will drift beyond determined surveillance criteria. Infrequent drift outside the surveillance criteria are expected. Excessive rack or sensor drift that is more than occasional may be indicative of more serious problems and should warrant further investigations.

ATTACHMENT 1
(Page 36 of 112)

Insert 3/4.3.1 and 3/4.3.2

TECHNICAL SPECIFICATION BASES

3/4.3.1 and 3/4.3.2 (Cont'd)

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) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) Feed water isolation, (4) Startup of the emergency diesel generators, (5) Containment spray pumps start and automatic valves position (6) Containment ventilation isolation, (7) Steam line isolation, (8) Turbine trip, (9) Auxiliary feedwater pumps start and automatic valves position, (10) Containment cooling fans start and automatic valves position, (11) Intake cooling water and component cooling water pumps start and automatic valves position, and (12) Control Room Isolation and Ventilation Systems start. This system also provides a feedwater system isolation to prevent SG overfill. Steam Generator overfill protection is not part of the Engineered Safety Features Actuation System (ESFAS), and is added to the Technical Specifications only in accordance with NRC Generic Letter 89-19.

Item 5 of Table 3.3-2 requires that two trains of feedwater isolation actuation logic and relays be OPERABLE in Modes ~~1 and 2~~. Operability requires:

1,2 and 3

Isolation of both the normal feedwater branch and the bypass branch lines during a safety injection actuation signal or high-high steam generator water level signal, and

Two independent trains of automatic actuation logic and actuation relays.

In the event that maintenance and/or in-service testing is required on a feedwater regulating valve in Mode ~~1 or 2~~, the above requirements can be met by closing the isolation valve upstream of the affected feedwater regulating valve, administratively controlling the position of the isolation valve, and controlling feedwater flow with an OPERABLE feedwater regulating valve (main or bypass).

When complying with ACTION 23 for Table 3.3-2 Functional Unit 6.d. the plant does not enter Limiting Condition for Operation (LCO) 3.0.3. ACTION 23, in the wording "comply with Specification 3.0.3", requires actions to be taken that are the same as those described in LCO 3.0.3, without any requirement to enter LCO 3.0.3. ACTION 23 has designated conditions under which the specific prescribed ACTIONS of within 1 hour action shall be initiated to place the unit, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours,

through automatic closure of the main feedwater and main feedwater bypass flow control valves (FCV) or automatic closure of the feedwater isolation valves (FIV).

The Steam Generator Pressure-Low and Steam Line Pressure- Low functions, items 1.f and 4.d of Table 3.3-3, are anticipatory in nature and have a typical lead/lag ratio of 50/5. The 50/5-second lead/lag function is needed to assure acceptable results for the Hot Full Power and Hot Zero Power steam line break analyses

INSERT FOR 3/4.3.1 AND 3/4.3.2

The trip setpoints used in the bistables for Functional Units 1f, 4d, 5c, and 6b are based on the analytical limits stated in WCAP-17070-P. The selection of these analytical limits is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49, the NTS's are conservative with respect to the analytical limits. A detailed description of the methodology used to determine the ESFAS NTS's, Allowable Value, as-left tolerance, and as found tolerance including their explicit uncertainties, is provided in WCAP-17070-P which incorporates all of the known uncertainties applicable to these functional units. The magnitudes of these uncertainties are factored into the determination of each ESFAS NTS and corresponding Allowable Value.

The NTS for Functional Units 1f, 4d, 5c, and 6b is the value at which the bistables are set and is the expected value to be achieved during calibration. The NTS value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable is considered to be properly adjusted when the as-left NTS value is within the as-left tolerance for CHANNEL CALIBRATION uncertainty allowance (i.e., ±rack calibration). The NTS value is therefore considered a "nominal value" (i.e., expressed as a value without inequalities) for the purposes of the COT and CHANNEL CALIBRATION. These functional Units have been modified by two notes as identified in Table 3.3-1. Note (a) requires evaluation of channel performance for the condition where the as found setting for the nominal trip setpoint is outside of the as found criterion. As stated above, these instances will be entered into the plant corrective action process and the channel will be evaluated to verify that it is functioning as required before returning the channel to service. Note (b) requires that the channel as left setting must be within the as left tolerance band. As noted above a channel is considered to be properly calibrated with the as left NTS value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e., ± rack calibration accuracy).

WCAP-17070-P methodology for determining analytical limits only applies to those ESFAS functions that support the safety analysis. Certain ESFAS functions such as the loss of voltage UV signal to the 480V load centers only provide equipment protection and therefore their analytical limit is not required to meet the WCAP-17070-P methodology.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.1 (Cont'd)

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 275°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 either: (1) Restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) 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. The 50°F limit includes instrument error.

The Technical Specifications for Cold Shutdown allow an inoperable RHR pump to be the operating RHR pump for up to 2 hours for surveillance testing to establish operability. This is required because of the piping arrangement when the RHR system is being used for Decay Heat Removal.

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 293,330 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an RCS vent opening of at least 2.50 square inches will provide overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Mitigating 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 first 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.

In Mode 5 only one pressurizer code safety is required for overpressure protection. In lieu of an actual operable code safety valve, an unisolated and unsealed vent pathway (i.e., a direct, unimpaired opening, a vent pathway with valves locked open and/or power removed and locked on an open valve) of equivalent size can be taken credit for as synonymous with an OPERABLE code safety.

The pressurizer safety valves are set to open at an RCS pressure of 2465 psig +2% and -3% to avoid exceeding the maximum design pressure safety limit and to maintain accident assumptions. The pressurizer safety valve lift setting is needed to assure acceptable results for the Loss of Load/ Turbine Trip analysis. The upper and lower pressure tolerance limits are based on the tolerance requirements assumed in the safety analyses.

ATTACHMENT 1
(Page 59 of 112)

TECHNICAL SPECIFICATION BASES

3/4.4.9 (Cont'd)

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location, the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

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 ~~III~~, Appendix G:

1996 Edition of the

XI

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures ~~3.4-2~~ ~~to~~ ~~3.4-4~~ for the service period specified thereon:

   3.4-2 and 3.4-3

   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

   3.4-2 and 3.4-3

   b. Figures ~~3.4-2 to 3.4-4~~ 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.

ATTACHMENT 1
(Page 60 of 112)

TECHNICAL SPECIFICATION BASES

WCAP-14040-NP-A, Revision 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves.

3/4.4.9 (Cont'd)

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 320°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 properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, the version of the ASTM E185 standard required by 10 CFR 50, Appendix H, and in accordance with additional reactor vessel requirements. *

XI 1996

The properties are then evaluated in accordance with Appendix G of the ~~1983~~ Edition of Section ~~III~~ of the ASME Boiler and Pressure Vessel Code and the additional requirements of 10 CFR 50, Appendix G and the calculation methods described in Westinghouse Report ~~GTSD-A-1.12, Procedure for Developing Heatup and Cooldown Curves~~.

48

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, ~~RTNDT~~, at ~~the end of 19~~ effective full power years (EFPY) of service life. The ~~19~~ EFPY service life period is chosen such that the limiting ~~RTNDT~~, at the 1/4T location in the core region is greater than the ~~RTNDT~~, of the limiting unirradiated material. The selection of such a limiting ~~RTNDT~~ assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

heatup rates of 60 and

RT_{NDT}

3.4-2 and 3.4-3,

The heatup and cooldown limit curves, Figures ~~3.4-2, 3.4-3 and 3.4-4~~ are composite curves prepared by determining the most conservative case with either the inside or outside wall controlling, for ~~any heatup rate up to~~ 100 degrees F per hour and cooldown rates of ~~up to~~ 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of predicted adjusted reference temperature at the end of the applicable service period (~~19~~ EFPY).

48

20,40,60, and

The reactor vessel materials have been tested to determine their initial ~~RTNDT~~; the results of these tests are shown in Tables B 3/4.4-1 and B 3/4.4-2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the ~~RTNDT~~. Therefore, an adjusted reference temperature, based upon the fluence and chemistry factors of the material has been predicted using Regulatory Guide 1.99, Revision 2, dated May 1988, Radiation Embrittlement of Reactor Vessel Materials. The heatup and cooldown limit curves of Figures ~~3.4-2, 3.4-3, and 3.4-4~~ include predicted adjustments for this shift in ~~RTNDT~~ at the end of the applicable service period.

3.4-2 and 3.4-3

RT_{NDT}

* Topical Report BAW-2308, Revision 2-A is the source for the initial weld materials properties for Linde 80 welds.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.9 (Cont'd)

The actual shifts in ~~RTNDT~~ RT_{NDT}, of the vessel materials will be established periodically during operation by removing and evaluating, in accordance with the version of the ASTM E185 standard required by 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel.

Since the limiting circumferential weld beltline material in Unit 3 is identical to the limiting circumferential weld beltline material in Unit 4 (Intermediate to Lower Shell Circumferential Weld), ~~Since the limiting beltline materials (Intermediate to Lower Shell Circumferential Weld) in Units 3 and 4 are identical,~~ the RV surveillance program was integrated and the results from capsule testing is applied to both Units. The surveillance capsule ~~T~~ results ~~from Unit 3 (WCAP 8631) and Unit 4 (SWRI 02 4221) and the capsule V results from Unit 3 (SWRI 06 8576) were used~~ were considered with the methodology in Regulatory Guide 1.99, Revision 2, to ~~provide limiting~~ determine material properties information for generating the heatup and cooldown curves in Figures ~~3.4-2, 3.4-3, and 3.4-4~~ 3.4-2 and 3.4-3. The integrated surveillance program along with similar identical reactor vessel design and operating characteristics allows the same heatup and cooldown limit curves to be applicable at both Unit 3 and Unit 4.

ATTACHMENT 1
(Page 62 of 112)

TECHNICAL SPECIFICATION BASES

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS (UNIT 3)

Component	Material Type	Cu (%)	Ni (%)	P (%)	NDTT (°F)	50 ft lb/35 mils Lateral Expansion Temp (°F) Long	Trans	RT_{NDT} (°F)	Minimum Upper Shelf (ft lb) Long	Trans
Cl. Hd. Dome	A302 Gr. B	-	-	0.010	0	-	36(a)	0	>70	> 45.5(a)
Cl. Hd. Flange	A508 Cl. 2	-	0.72	0.010	~~44(a)~~ -	-	31(a)	~~44~~ -50	>118	> 76.5(a)
Ves. Sh. Flange	A508 Cl. 2	-	0.65	0.010	-23(a)	-	-41(a)	~~2?~~ -23	>120	> 78(a)
Inlet Nozzle 1	A508 Cl. 2	- 0.16(b)	0.76	0.019	60(a)	-	NA	60	NA	~~NA~~ 109(c)
Inlet Nozzle 2	A508 Cl. 2	- 0.16(b)	0.74	0.019	60(a)	-	NA	60	NA	~~NA~~ 109(c)
Inlet Nozzle 3	A508 Cl. 2	- 0.16(b)	0.80	0.019	60(a)	-	NA	60	NA	~~NA~~ 109(c)
Outlet Nozzle 1	A508 Cl. 2	- 0.16(b)	0.79	0.010	27(a)	-	9(a)	27	>110	>71.5(a)
Outlet Nozzle 2	A508 Cl. 2	- 0.16(b)	0.72	0.010	7(a)	-	-22(a)	7	>111	>72(a)
Outlet Nozzle 3	A508 Cl. 2	- 0.16(b)	0.72	0.010	42(a)	-	23(a)	42	>140	>91(a)
Upper Shell	A508 Cl. 2	- 0.11	0.68	0.010	50	-	44(a)	50	~~>129~~ -	~~>83.5(a)~~ 99
Inter. Shell	A508 Cl. 2	~~0.058~~ 0.06	0.70	0.010	40	-	25(a)	40	~~>122~~ -	~~>79(a)~~ 93
Lower Shell	A508 Cl. 2	~~0.079~~ 0.08	0.67	0.010	30	-	2(a)	30	~~163~~ -	~~106(a)~~ 100
Trans. Ring	A508 Cl. 2	-	0.69	0.013	60(a)	-	58(a)	60	>109	>70.5(a)
Bot. Hd. Dome	A302 Gr. B	-	-	0.010	-10	-	NA	30	NA	NA
Inter. to Lower Shell Girth Weld	~~SAW~~ LINDE 80	~~0.26~~ 0.23	~~0.60~~ 0.59	0.011	~~10(b)~~ -	-	63	~~10(b)~~ -53.5	-	~~63~~ 65
HAZ	HAZ	-	-	-	0(a)	-	0	0	-	168
Upper to Inter. Shell Girth Weld	LINDE 80	0.26	0.60	-	-	-	-	-33.2	NA	70(c)

(a) Estimated values based on NUREG-0800, Branch Technical Position - MTEB 52

~~(b) Actual Value~~

(b) Conservative values based on Oak Ridge National Laboratory document ORNL/TM-2006/530

(c) Generic values based on BAW-2313, Revision 6, AREVA NP Document No. 77-2313-006, November 2008

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

TABLE B 3/4.4-2

REACTOR VESSEL TOUGHNESS (UNIT 4)

Component	Material Type	Cu (%)	Ni (%)	P (%)	NDTT (°F)	50 ft lb/35 mils Lateral Expansion Temp (°F) Long	Trans	RT_{NDT} (°F)	Minimum Upper Shelf (ft lb) Long	Trans
Cl. Hd. Dome	A302 Gr. B	-	-	0.008	-20	-	NA	30	NA	NA
Cl. Hd. Flange	A508 Cl. 2	-	0.72	0.010	~~-4(a)~~ -	-	27(a)	~~-4~~ -50	199	129(a)
Ves. Sh. Flange	A508 Cl. 2	-	0.68	0.010	-1(a)	-	-11(a)	-1	176	114(a)
Inlet Nozzle 1	A508 Cl. 2	0.08	0.71	0.009	60(a)	-	NA	60	NA	~~NA~~ 109(c)
Inlet Nozzle 2	A508 Cl. 2	- 0.16(b)	0.84	0.019	60(a)	-	NA	60	NA	~~NA~~ 109(c)
Inlet Nozzle 3	A508 Cl. 2	- 0.16(b)	0.75	0.008	16(a)	-	13(a)	16	162	105(a)
Outlet Nozzle 1	A508 Cl. 2	- 0.16(b)	0.78	0.010	7(a)	-	-25(a)	7	165	107(a)
Outlet Nozzle 2	A508 Cl. 2	- 0.16(b)	0.68	0.010	38(a)	-	16(a)	38	160	104(a)
Outlet Nozzle 3	A508 Cl. 2	- 0.16(b)	0.70	0.010	60(a)	-	42(a)	60	143	93(a)
Upper Shell	A508 Cl. 2	- 0.11	0.70	0.010	40	-	32(a)	40	~~156~~ -	~~101(a)~~ 103
Inter. Shell	A508 Cl. 2	~~0.054~~ 0.05	~~0.69~~ 0.68	0.010	50	-	90(a)	50	~~143~~ -	~~93(a)~~ 88
Lower Shell	A508 Cl. 2	~~0.056~~ 0.06	0.74	0.010	40	-	38(a)	40	~~149~~ -	~~97(a)~~ 86
Trans. Ring	A508 Cl. 2	-	0.69	0.011	60(a)	-	30(a)	60	NA	NA
Bot. Hd. Dome	A302 Gr. B	-	-	0.010	10	-	30(a)	10	NA	NA
Inter. to Lower Shell Girth Weld	~~SAW~~ LINDE 80	~~0.26~~ 0.23	~~0.60~~ 0.59	0.011	~~10(b)~~ -	-	63	~~10(b)~~ -53.5	NA	~~63~~ 65
HAZ	HAZ	-	-	-	0	-	NA	0	NA	140

(a) Estimated values based on NUREG-0800, Branch Technical Position - MTEB 52

~~(b) Actual Value~~

(b) Conservative values based on Oak Ridge National Laboratory document ORNL/TM-2006/530

(c) Generic values based on BAW-2313, Revision 6, AREVA NP Document No. 77-2313-006, November 2008

(d) Value based on BAW-1910P, B&W Document No. 77-1164769-00, August 1986

Component	Material Type	Cu (%)	Ni (%)	P (%)	NDTT (°F)	Long	Trans	RT_{NDT} (°F)	Long	Trans
Upper to Inter. Shell Girth Weld (Inner 67%)	LINDE 80	0.26	0.60	-	-	-	-	-33.2	NA	72(d)
Upper to Inter. Shell Girth Weld (Outer 33%)	LINDE 80	0.32	0.58	-	-	-	-	-31.1	NA	NA

W2003:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.9 (Cont'd)

Section XI of the 1996 Edition

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in ~~Section III~~ of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and Westinghouse Report ~~GTSD-A-1.12, Procedure for Developing Heatup and Cooldown Curves.~~

Appendix G of the 1996 Edition of ASME Section XI

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 a semi-elliptical 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. The dimensions of this postulated crack, referred to in ~~Appendix G of ASME Section III~~ as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this 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, ~~RTNDT~~, is used and this includes the radiation-induced shift, ~~ARTNDT~~, corresponding to the end of the period for which heatup and cooldown curves are generated.

ΔRT_{NDT} K_I RT_{NDT}

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, ~~KI~~, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, ~~KIR~~, for the metal temperature at that time. ~~KIR~~ is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The ~~KIR~~ curve is given by the equation:

K_{Ia}

K_{Ia} RT_{NDT}

$$\cancel{K_{IR}} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: ~~KIR~~ is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature ~~RTNDT~~. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$< K_{Ia}$

$$C\,K_{IM} + K_{IT} \cancel{\leq} \cancel{K_{IR}} \quad (2)$$

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

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

K_{Ia}

$\cancel{K_{IR}}$ = 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 hydrostatic and leak test operations.

ATTACHMENT 1
(Page 65 of 112)

TECHNICAL SPECIFICATION BASES

3/4.4.9 (Cont'd)

At any time during the heatup or cooldown transient, ~~KIR~~ $\boxed{K_{Ia}}$ is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for ~~RTNDT~~ $\boxed{RT_{NDT}}$, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, ~~KIT~~ $\boxed{K_{IT}}$, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

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. During cooldown, the controlling location of the flaw is always at the inside of the wall 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 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 vessel location is at a higher temperature than the fluid adjacent to the vessel ID. 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 ~~KIR~~ $\boxed{K_{Ia}}$ at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in ~~KIR~~ $\boxed{K_{Ia}}$ exceeds ~~KIT~~ $\boxed{K_{IT}}$, 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.

ATTACHMENT 1
(Page 66 of 112)

TECHNICAL SPECIFICATION BASES

3/4.4.9 (Cont'd)

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 of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the ~~KIR~~ K_{Ia} for the 1/4T crack during heatup is lower than the ~~KIR~~ K_{Ia} 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 ~~KIR's~~ K_{Ia} 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 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~~ rates, 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~~ situation.

Finally, the 10 CFR 50 Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. The rule states that the minimum metal temperature for the flange regions should be at least 120°F higher than the ~~limiting RTNDT~~ limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (621 psig). Since the ~~limiting RTNDT~~ limiting RT_{NDT} for the flange regions for Turkey Point Units 3 and 4 is ~~44°F~~ -1°F, the minimum temperature required for pressure of 621 psig and greater based on the Appendix G rule is ~~164°F~~ 119°F. The heatup and cooldown curves as shown in Figures ~~3.4-2 to 3.4-4~~ 3.4-2 and 3.4-3 clearly satisfy the above requirement by ample margins.

ATTACHMENT 1
(Page 77 of 112)

TECHNICAL SPECIFICATION BASES

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 of 2.5 psig with respect to the outside atmosphere, and (2) The containment peak pressure does not exceed the design pressure of 55 psig during LOCA conditions.

~~The LOCA containment integrity analysis determines a peak pressure of 48.3 psig for those cases performed with an initial containment pressure of +0.3 psig (represents nominal containment pressure conditions). This analysis confirms that the containment pressure will not exceed the limit of 49.9 psig assumed for containment leak rate testing. The analysis cases performed at an initial containment pressure of +3.0 psig result in a maximum peak pressure of 51.5 psig, which confirms that the containment pressure will not exceed the design value of 55 psig.~~

3/4.6.1.5 Air Temperature

The limitations on containment average air temperature ensure that the design limits for a LOCA are not exceeded, and that the environmental qualification of equipment is not impacted. If temperatures exceed 120°F, but remain below 125°F for up to 336 hours during a calendar year, no action is required. If the 336-hour limit is approached, an evaluation may be performed to extend the limit if some of the hours have been spent at less than 125°F. 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 analyzed peak pressure of 49.9 psig in the event of a LOCA.~~ The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

Some containment tendons are inaccessible at one end due to personnel safety considerations at potential steam exhaust locations. These tendons, if selected for examination, will be exempted from the full examination requirements, and the following alternative examinations shall be performed:

will be required to ensure that the containment will withstand the design pressure of 55 psig in the event of a LOCA.

1. The accessible end of each exempt tendon shall be examined in accordance with IWL-2524 and IWL-2525.

2. For each exempt tendon, a substitute tendon shall be selected and examined in accordance with IWL requirements.

3. In addition, an accessible tendon located as close as possible to each exempt tendon shall be examined at both ends in accordance with IWL-2524 and IWL-2525.

TECHNICAL SPECIFICATION BASES

3/4.7 Plant Systems

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% (1193.5 psig) of its design pressure of 1085 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).

Replace with 3/4.7.1.1 Insert A on following page

3/4.7.1.1 Safety Valves

~~The specified valve lift settings and relieving capacities are in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 10,670,000 lbs/h which is 111% of the total secondary steam flow of 9,600,000 lbs/h at 100% RATED THERMAL POWER. A minimum of one 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:

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

Where:

Hi Φ = Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP

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

K = Conversion factor; 947.82 (Btu/sec)/Mwt

3/4.7.1.1 Insert A

The primary purpose of the main steam safety valves (MSSVs) is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Four MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 10.2. The MSSVs must have sufficient capacity to limit the secondary system pressure to less than or equal 110 percent of the steam generator design pressure in order to meet the requirements of ASME Code, Section III. The total relieving capacity for all valves on all of the steam lines is 9.936×10^6 lbs/hr which is 85.27 percent of the total secondary steam flow of 11.65×10^6 lbs/hr at 100% RATED THERMAL POWER. The MSSV design includes staggered setpoints, according to Table 3.7-2 in the accompanying limiting condition for operation (LCO), so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.7.1.2 (Cont'd)

ACTION statement 2 describes the actions to be taken when both auxiliary feedwater trains are inoperable. The requirement to verify the availability of both standby feedwater pumps is to be accomplished by verifying that both pumps have successfully passed their monthly surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If no alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke ACTION statement 1 for the other train. If both standby feedwater pumps are made available before one auxiliary feedwater train is returned to an OPERABLE status, then the affected units shall be placed in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

ACTION statement 3 describes the actions to be taken when a single auxiliary feedwater pump is inoperable. The requirement to verify that two independent auxiliary feedwater trains are OPERABLE is to be accomplished by verifying that the requirements for Table 3.7-3 have been successfully met for each train within the last surveillance interval. The provisions of Specification 3.0.4 are not applicable to the third auxiliary feedwater pump provided it has not been inoperable for longer than 30 days. This means that a units can change OPERATIONAL MODES during a unit's heatup with a single auxiliary feedwater pump inoperable as long as the requirements of ACTION statement 3 are satisfied.

The specified flow rate acceptance criteria conservatively bounds the limiting AFW flow rate modeled in the single unit loss of normal feedwater analysis. Dual unit events such as a two unit loss of offsite power require a higher pump flow rate, but it is not practical to test both units simultaneously. The monthly flow surveillance test specified in 4.7.1.2.1.1 is considered to be a general performance test for the AFW system and does not represent the limiting flow requirement for AFW. Check valves in the AFW system that require full stroke testing under limiting flow conditions are tested under Technical Specification 4.0.5.

The monthly testing of the auxiliary feedwater pumps will verify their operability. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps.

3/4.7.1.3 Condensate Storage Tank

each with a capacity of 250,000 gallons

There are two (2) seismically designed ~~250,000~~ gallons condensate storage tanks. A minimum indicated volume of 210,000 gallons is maintained for each unit in MODES 1, 2 or 3. The OPERABILITY of the condensate storage tank with the minimum indicated volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately ~~23~~ hours or maintain the Reactor Coolant System at HOT STANDBY conditions for ~~15~~ hours and ~~then~~ cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed in operation.

18

4

9 hours to

which provides margin over the analysis minimum required volume of 207,637 gallons per PTN-BFJM-95-008

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.7.1.5 (Cont'd)

The Main Steam Bypass Valves (MSBV) as motor operated valves are required to provide the capability to warm the main steam lines and to equalize the steam pressure across the associated Main Steam Isolation Valve (MSIV). The MSBVs are provided with a motor operator to close on a main steam isolation signal if open. The MSIVs and their associated MSBVs are not Containment Isolation Valves. The MSBVs are not covered in any Technical Specifications and no LCO or Action Statements apply to them.

3/4.7.1.6 Standby Steam Generator Feedwater System

The purpose of this specification and the supporting surveillance requirements is to assure operability of the non-safety grade Standby Steam Generator Feedwater System. The Standby Steam Generator Feedwater System consists of commercial grade components designed and constructed to industry and FPL standards of this class of equipment located in the outdoor plant environment typical of FPL facilities system wide. The system is expected to perform with high reliability, i.e., comparable to that typically achieved with this class of equipment. FPL intends to maintain the system in good operating condition with regard to appearance, structures, supports, component maintenance, calibrations, etc.

The function of the Standby Feedwater System for OPERABILITY determinations is that it can be used as a backup to the Auxiliary Feedwater (AFW) System in the event the AFW System does not function properly. The system would be manually started, aligned and controlled by the operator when needed.

The A pump is electric-driven and is powered from the non-safety related C bus. In the event of a coincident loss of offsite power, the B pump is diesel driven and can be started and operated independent of the availability of on-site or off-site power.

A supply of ~~65,000~~ 77,000 gallons from the Demineralized Water Storage Tank for the Standby Steam Generator Feedwater Pumps is sufficient water to remove decay heat from the reactor for six (6) hours for a single unit or two (2) hours for two units. This was the basis used for requiring ~~65,000~~ 77,000 gallons of water in the non-safety grade Demineralized Water Storage Tank and is judged to provide sufficient time for restoring the AFW System or establishing make-up to the Demineralized Water Storage Tank.

The minimum indicated volume (~~135,000~~ 145,000 gallons) consists of an allowance for level indication instrument uncertainties (approximately 15,000 gallons) for water deemed unusable because of tank discharge line location and vortex formation (approximately ~~50,300~~ 50,200 gallons) and the minimum usable volume (~~65,000~~ 77,000 gallons). The minimum indicated volume corresponds to a water level of ~~8.5~~ 9.2 feet in the Demineralized Water Storage Tank.

The Standby Steam Generator Feedwater Pumps are not designed to NRC requirements applicable to Auxiliary Feedwater Systems and not required to satisfy design basis events requirements. These pumps may be out of service for up to 24 hours before initiating formal notification because of the extremely low probability of a demand for their operation.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.7.1.6 (Cont'd)

The guidelines for NRC notification in case of both pumps being out of service for longer than 24 hours are provided in applicable plant procedures, as a voluntary 4-hour notification.

Adequate demineralized water for the Standby Steam Generator Feedwater system will be verified once per 24 hours. The Demineralized Water Storage Tank provides a source of water to several systems and therefore, requires daily verification.

3/4.7.1.6 Standby Steam Generator Feedwater System

The Standby Steam Generator Feedwater Pumps will be verified OPERABLE monthly on a STAGGERED TEST BASIS by starting and operating them in the recirculation mode. Also, during each unit's refueling outage, each Standby Steam Generator Feedwater Pump will be started and aligned to provide flow to the nuclear unit's steam generators.

This surveillance regimen will thus demonstrate operability of the entire flow path, backup non-safety grade power supply and pump associated with a unit at least each refueling outage. The pump, motor driver, and normal power supply availability would typically be demonstrated by operation of the pumps in the recirculation mode monthly on a staggered test basis.

The diesel engine driver for the B Standby Steam Generator Feedwater Pump will be verified operable once every 31 days on a staggered test basis performed on the B Standby Steam Generator Feedwater Pump. In addition, an inspection will be performed on the diesel at least once every 18 months in accordance with procedures prepared in conjunction with its manufacture's recommendations for the diesel's class of service. This inspection will ensure that the diesel driver is maintained in good operating condition consistent with FPLs overall objectives for system reliability.

Insert 3/4.7.1.7

3/4.7.2 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 active failure, is consistent with the assumptions used in the safety analyses. One pump and two heat exchangers provide the heat removal capability for accidents that have been analyzed.

3/4.7.3 Intake Cooling Water System

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

Insert 3/4.7.1.7

3/4.7.1.7 Feedwater Isolation

The Feedwater Control Valves (FCVs) isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The function of the non-safety-grade Feedwater Isolation Valves (FIVs) is to provide the second isolation of MFW flow to the secondary side of the steam generators following a HELB. Closure of the FCVs, FIVs, and the tandem bypass line valves terminate flow to the steam generators for feedwater line breaks (FWLBs) occurring upstream of the FCVs or FIVs.

The LCO requires Main Feedwater Isolation be OPERABLE. Main Feedwater Isolation consists of the three FCVs, three FIVs, six bypass line isolation valves, and their associated isolation circuits. This LCO ensures that in the event of an SLB inside containment, a single failure cannot result in continued MFW flow into the containment. Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB inside containment.

In MODES 1, 2, and 3, the FCVs and FIVs and the bypass lines valves are required to be OPERABLE to limit the amount of available fluid added to containment in a SLB inside containment. In MODES 4, 5, and 6, steam generator energy is low and FW isolation is not required.

The Action statements are modified by a Note indicating that separate Condition entry is allowed for each valve.

With one FCV, FIV or bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE FIV and tandem bypass line isolation valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable FCVs and FIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

With two valves in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the FCV or FIV, or otherwise isolate the affected flow path.

SR 4.7.1.7.a verifies that each FCV, FIV, and bypass line valve will actuate to its isolation position on an actuation or simulated actuation signal. The 18 month Frequency is based on a refueling cycle interval and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

SR 4.7.1.7.b verifies that the closure time of each FCV, FIV, and bypass line valve, when tested in accordance with the Inservice Testing Program, is within the limits assumed in the accident and containment analyses. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code Section XI (Ref. 3), quarterly stroke requirements during operation in MODES 1 and 2.

ATTACHMENT 1
(Page 111 of 112)

TECHNICAL SPECIFICATION BASES

500

3/4.9.14 Spent Fuel Storage

Replace with INSERT FOR TS BASES 3/4.9.14

The spent fuel storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure: a) Keff ≤0.95 with a minimum soluble boron concentration of ~~650~~ ppm present, and b) Keff <1.0 when flooded with unborated water for normal operations and postulated accidents.

~~The spent fuel racks are divided into two regions. Region I racks have a 10.6 inch center-to-center spacing and Region II racks have a 9.0 inch center-to-center spacing. Because of the larger center-to-center spacing and poison (B10) concentration of Region I cells, the only restriction for placement of fuel is that the initial fuel assembly enrichment is equal to or less than 4.5 weight percent of U-235. The limiting value of U-235 enrichment is based upon the assumptions in the spent fuel safety analyses and assures that the limiting criteria for criticality is not exceeded. Prior to placement in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.9-1, placement in a Region II cell is authorized. These positive controls assure that fuel enrichment limits assumed in the safety analyses will not be exceeded.~~

3/4.10 Special Test Exceptions

3/4.10.1 Shutdown Margin

This special test exception provides that a minimum amount of 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 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 measure control rod worth.

The 500 ppm value is needed to assure keff < 0.95 for normal operating conditions (the Boraflex remedy requirement). The criticality analysis needs 1700 ppm to assure keff < 0.95 under the worst case accident condition. There is significant margin between the calculated ppm requirement and the spent fuel boron concentration requirement of 2300 ppm. The higher boron concentration value is chosen because, during refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass.

INSERT FOR TS BASES 3/4.9.14 SPENT FUEL STORAGE

The spent fuel racks are divided into two regions, Region I and Region II. The Region I permanent racks have a 10.6 inch center-to-center spacing. The Region I cask area rack has a nominal 10.1 inch center to center spacing in the east-west direction and a nominal 10.7 inch center-to-center spacing in the north-south direction. The Region II racks have a 9.0 inch center-to-center spacing.

Any fuel for use at Turkey Point, and enriched to less than or equal to 5.0 wt% U-235, may be stored in the cask area rack. The restrictions for placement of fresh fuel in the permanent Region I spent fuel racks are that the initial fuel assembly enrichment is equal to or less than 4.7 weight percent of U-235 with no IFBA rods or 5.0 weight percent of U-235 with at least 16 IFBA rods (or an equivalent amount of another burnable absorber) and that the fresh fuel either contains a full length rod control cluster assembly (RCCA) or be in a checkerboard array of fuel and empty cells. The limiting value of U-235 enrichment and the requirement for full length RCCA or a checkerboard array of empty cells are based upon the assumptions in the criticality analysis and assures that the limiting criteria for criticality are not exceeded.

Irradiated fuel will either be “pre-EPU” or “EPU” fuel. A pre-EPU fuel assembly is a fuel assembly that was never in the core at the Extended Power Uprate (EPU) condition. An EPU fuel assembly is a fuel assembly that is irradiated in the core for any amount of time at the EPU condition. Prior to placement of irradiated fuel in Region I or II spent fuel storage rack cell locations, strict controls are employed to evaluate burnup of the fuel assembly. Upon determination that the fuel assembly meets the nominal burnup requirements of either Table 5.5-1, Table 5.5-2, Table 5.5-4, or Table 5.5-5 (depending on whether the fuel assembly and surrounding neighbors are pre-EPU or EPU), placement in a Region I or II cell in accordance with analyzed storage arrangements is authorized. For all assemblies with blanketed fuel, the initial enrichment is based on the central zone enrichment (i.e., between the axial blankets) consistent with the assumptions of the analysis. These positive controls assure that fuel enrichment limits, burnup, and post-irradiation cooling time requirements assumed in the criticality safety analyses will not be violated.