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2CAN050901

May 13, 2009

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

SUBJECT: License Amendment Request to Revise the Departure From Nucleate Boiling Ratio (DNBR) Safety Limit
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

REFERENCES:

1. WCAP-16523-P-A, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," August 2007

2. WCAP-16500-P-A, "CE 16 x 16 Next Generation Fuel Core Reference Report," Revision 0, August 2007

3. Entergy Letter to the US Nuclear Regulatory Commission, Attn: Document Control Desk from Mr. T. G. Mitchell, Vice President, Arkansas Nuclear One, "Technical Specification Change to Modify RCS Flow Verification," dated November 13, 2008

4. WCAP-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel," Revision 0, May 2000

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed change will modify the Technical Specification (TS) 2.1.1.1 DNBR safety limit based upon the Combustion Engineering (CE) 16 x 16 Next Generation Fuel (NGF) design and the associated Departure from Nucleate Boiling (DNB) correlations (References 1 and 4). The limiting safety system setting reactor trip setpoint for DNBR – Low, listed in TS Table 2.2-1, is not impacted by this change.

The NGF design improves fuel reliability to resolve grid-to-rod fretting failures, improves fuel performance for high duty operation, and provides enhanced DNB margin. The licensing basis for this fuel design is provided in Reference 2. As noted in Reference 2, the ABB-TV DNB correlation gives conservative predictions relative to the NGF DNB test data. As a result of this NGF test data, a new DNB correlation for the NGF fuel assembly was submitted to the NRC. The NRC has approved the new correlation for safety and setpoint analyses (Reference 1). This proposed change is to make the TS consistent with that approval and application of the new correlation. Justification for the proposed change to TS 2.1.1.1 is provided in Attachment 1. A mark-up of the affected TS is provided in Attachment 2.

References 1, 2, and 4 cited above and all but one of the references provided in Attachment 1 are currently part of the ANO-2 licensing basis.

A mark-up of the associated TS 2.1.1.1 Bases is provided in Attachment 3 for information only. The TS bases, controlled in accordance with the TS Bases Control Program of TS 6.5.14, will be revised accordingly upon approval of this request.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using the criteria in 10 CFR 50.92(c), and it has been determined that the changes involve no significant hazards consideration. The bases for these determinations are included in the attached submittal.

The proposed change does not include any new commitments.

This submittal does not impact the Reference 3 submittal.

Entergy requests approval of the proposed amendment by September 16, 2009, to support the Fall 2009 refueling outage. Once approved, the amendment shall be implemented after the current cycle (Cycle 20) is completed and prior to the operation of Cycle 21.

If you have any questions or require additional information, please contact David Bice at 479-858-5338.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 13, 2009.

Sincerely,

Original signed by K. Walsh

KTW/rwc

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Proposed Technical Specification Bases Changes (mark-up) – For Information

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Attachment 1

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Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2).

The proposed change will modify Technical Specification (TS) 2.1.1.1 to account for the Combustion Engineering (CE) 16 x 16 Next Generation Fuel (NGF) and different NRC reviewed and approved Departure from Nucleate Boiling (DNB) correlations (References 7.1 and 7.3). These new correlations will be implemented in the safety analyses for the next fuel cycle of operation consistent with NRC approved methodologies.

2.0 PROPOSED CHANGE

The safety limit for the Departure from Nucleate Boiling Ratio (DNBR) is revised from greater than or equal to 1.25 with the CE-1 correlation to greater than or equal to 1.23 with the WSSV-T and ABB-NV correlations. The DNBR value of 1.23 is the more limiting value that was determined using either the WSSV-T or the ABB-NV correlation.

A draft mark-up of the associated TS Bases is provided in Attachment 3 for information only. The TS bases, controlled in accordance with the TS Bases Control Program of TS 6.5.14, will be revised accordingly upon approval of this request.

3.0 BACKGROUND

The goals for the NGF assembly design include improving fuel reliability to resolve grid-to-rod fretting failures, improving fuel performance for high duty operation, and providing enhanced thermal margin for DNB. The NGF design improves heat transfer performance of the fuel design through the following design changes: (1) the addition of Intermediate Flow Mixer (IFM) grids in the fuel assembly and (2) the addition of side-supported mixing vanes on both the Mid grids and IFM grids (Reference 7.2).

The new design features of the NGF assembly for thermal improvements have been verified with respect to applicable thermal hydraulics design criteria through testing and analysis. The ABB-TV correlation (Reference 7.3) developed for TURBO fuel was demonstrated to be conservative for the NGF test data. The TURBO fuel does not have all of the design features, such as the IFM grids, as the NGF design. To more accurately reflect the thermal performance of the NGF assembly, a new DNB correlation was developed. This new correlation was entitled WSSV. The WSSV correlation is used with the thermal hydraulic code VIPRE and WSSV-T is used with the TORC thermal hydraulic code (Reference 7.1). This correlation is used in the mixing vane region of the core. The ABB-NV correlation is used to calculate the DNBR values in the hot channels in the non-mixing vane region of the core (Reference 7.3).

The WSSV-T correlation is described in Reference 7.1. The WSSV-T correlation coefficients were derived with the Westinghouse TORC (Reference 7.4) subchannel computer code. As described in Reference 7.1, the WSSV-T correlation is valid for use with the Westinghouse thermal hydraulic codes TORC and CETOP-D.

Specified fuel design limits are not to be exceeded during steady state operation, normal operational transients, and anticipated operating occurrences (AOOs). One of the ways this is accomplished is by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur.

Operation beyond the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in the heat transfer coefficient. Because of the steam film, high cladding temperatures may be reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

DNB is not an observable parameter during reactor operation. The observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) or DNB correlation. The CE-1 (Reference 7.5), ABB-NV (Reference 7.3) and WSSV-T (Reference 7.1) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The CE-1 correlation applies to the CE standard fuel. The ABB-NV correlation applies to both the standard CE fuel and the non-mixing vane regions of the NGF design. The WSSV-T correlation applies to the mixing vane regions of the NGF fuel assembly design. The local DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and AOOs is limited to 1.25 for the CE-1 correlation. The bounding value for the WSSV-T and the ABB-NV correlations is 1.23. This value includes allowances for the items listed in Table 3-1 of Reference 7.6.

The Reactor Protective System (RPS) is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and thermal power level that would result in a violation of the reactor core safety limits.

The Core Protection Calculator System (CPCS) is part of the RPS. It consists of four Core Protection Calculators (CPCs) and two Control Element Assembly Calculators (CEACs). The CPCS initiates the low DNBR and high Local Power Density (LPD) trips of the RPS in order to assure that fuel design limits on DNBR and fuel centerline temperatures are not exceeded during AOOs and to assist the Engineered Safety Features Actuation System (ESFAS) in limiting the consequences of certain postulated accidents. Each CPC channel receives safety grade sensor inputs and uses these inputs to calculate DNBR, LPD, and other parameters. The CEACs receive safety grade Control Element Assembly (CEA) position inputs and provide single CEA position-related factors to each CPC channel such that the CPCs respond appropriately to single CEA-related AOOs which require CPC protection.

The Core Operating Limit Supervisory System (COLSS) is a digital computer based on-line monitoring system that is used to issue alarm signals to the plant computer and to provide information to aid the operator in complying with TS operating limits on DNBR, total core power, peak linear heat rate (LHR), axial shape index (ASI), and azimuthal power tilt.

The CPCS and COLSS include thermal hydraulic algorithms derived from the thermal hydraulic design code CETOP-D. The CPCs continuously calculate DNBR and compare it to a trip setpoint in order to decide whether to issue a trip signal through the RPS. COLSS continuously calculates the DNB power operating limit (POL) and compares it to the calculated power in order to decide whether to issue an alarm signal to the plant computer.

The setpoint analysis is performed every reload cycle in order to calculate addressable constants for the CPCs and COLSS. Addressable constants are coefficients of the CPCs and COLSS algorithms which can be changed readily during startup or operation. Addressable constants include calibration coefficients, measurement results, uncertainty factors, adjustment factors, time delays, and trip setpoints. The primary purpose of the cycle specific setpoint analysis is to calculate the CPCs and COLSS power uncertainty factors, BERR1 and EPOL, respectively, that are used in the DNBR calculations.

Further information with regard to DNB is available in the ANO-2 Safety Analysis Report (SAR), Section 4.4.2.3.

4.0 TECHNICAL ANALYSIS

ANO-2 currently has a transitional core, a partial core of NGF fuel assemblies and the remaining portion the standard CE fuel design. During this transition, ANO-2 has not taken full advantage of the enhanced operating margin that is present in the NGF design. Following the refueling outage scheduled for the Fall of 2009, ANO-2 will have a complete core of NGF fuel assemblies. It is at this time that the full advantage of the DNB benefit of the mixing vanes, resulting in improved operating margin, will be realized. Since the NGF WSSV-T and ABB-NV DNB correlations will be used in the safety analyses for the reload, the safety limit provided in TS 2.1.1.1 is being updated to be consistent with these NRC-approved models.

The TORC code is used in reloads to perform detailed modeling of the core and hot assembly and to determine minimum DNBR in the hot assembly. The CETOP-D code is used in reload analyses to calculate the minimum DNBR in the hot subchannel. While the TORC code can be applied directly to the reload analyses, typically the TORC code is used to benchmark the CETOP-D DNBR results such that the CETOP-D results are conservative relative to TORC. The WSSV-T and ABB-NV correlations are in both TORC and CETOP-D. Therefore, the application of WSSV-T and ABB-NV correlations with CETOP-D is equivalent to their application with TORC (Reference 7.1).

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR safety limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR safety limit, uncertainties in the CHF correlation and system parameter uncertainties are statistically combined (References 7.6 and 7.9) to determine a DNBR safety limit. In the case of the NGF assembly design, two CHF correlations are used. As such the bounding value for DNBR determined by using each correlation in the process described above is the DNBR safety limit. This statistical limit protects the respective CHF safety limit. Additional retained thermal margin may also be applied to the statistical DNBR safety limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur during normal operation and AOOs.

The DNBR limits for the WSSV-T CHF correlation is 1.12 (Reference 7.1) and 1.13 for the ABB-NV correlation (Reference 7.3). The correlation uncertainties that form the basis for the more limiting value are combined with system parameter uncertainties using the methodology in References 7.6 and 7.9 to yield a DNBR safety limit of 1.23.

This change does not impact the system parameters or the computer code uncertainties. The change in the DNBR limit is due to the CHF correlation used.

The WSSV-T and ABB-NV correlations are used in the safety and setpoint analyses. However, because of existing hardware limitations, the CPC algorithm will retain the CE-1 correlation. Since the CPC thermal-hydraulic algorithm retains the CE-1 correlation, the DNBR-Low trip setpoint and Allowable Value remains set at 1.25. The CPC power uncertainty factor used in the DNBR calculations (addressable constant BERR1), is calculated using the WSSV-T and ABB-NV correlations in accordance with the setpoint methodology of Reference 7.6. The BERR1 constant is calculated such that a CPC trip at a DNBR of 1.25 using the CE-1 CHF correlation in the CPCs assures that the bounding DNBR safety limit of 1.23 for the WSSV-T and ABB-NV correlations will not be violated during normal operations and AOOs to at least a 95/95 probability / confidence level. Thus, the trip setpoint listed in TS Table 2.2-1 conservatively remains at 1.25 even though credit is being obtained (via the adjusted BERR1 constant) for the improved WSSV-T and ABB-NV correlations and the DNBR limit of 1.23. Therefore, compliance with 10 CFR 50.36 is maintained, as discussed in Section 5.1 below.

Based on the above, Entergy has determined that the proposed change is acceptable, maintains applicable regulatory requirements, and presents a no significant impact to nuclear or public safety.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.36, Technical Specifications, defines a safety limit as a limit upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.

General Design Criteria (GDC) 10, Reactor Design, requires that specified fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur.

The revised bounding Departure from Nucleate Boiling Ratio (DNBR) limit for NGF fuel using the WSSV-T and ABB-NV critical heat flux (CHF) correlations continues to meet the requirements for safety limits as defined by 10 CFR 50.36 and maintains the GDC 10 requirements.

10 CFR 50.36 also states “A Limiting Safety System Setting is the setting for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.” The process outlined in Reference 7.7 ensures that the trip setpoint of 1.25 listed in TS Table 2.2-1 will satisfy this requirement.

Based on the considerations discussed in Section 4.0 above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or the health and safety of the public.

References 7.1 through 7.6 have been reviewed and approved by the NRC and are part of the current licensing basis for Arkansas Nuclear One, Unit 2 (ANO-2). As such the limits and conditions listed in the safety evaluations for each of these references have previously been addressed by ANO-2. These limits and conditions remain in effect including the one in which a 6% operating margin penalty will be applied until Reference 7.7 has been reviewed and approved by the NRC. The draft NRC Safety Evaluation for this revision is scheduled to be issued no later than October 16, 2009.

In conclusion, Entergy has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any GDC differently than described in the Safety Analysis Report (SAR).

5.2 No Significant Hazards Consideration

The proposed change will modify the Departure from Nucleate Boiling Ratio DNBR safety limit of Technical Specification (TS) 2.1.1.1, DNBR, to a lower safety limit value for Arkansas Nuclear One, Unit 2 (ANO-2).

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   No changes to plant equipment or operating procedures are required due to the change in the safety limit for DNBR. This change does not impact any of the accident initiators. The analyses of the reload are performed using NRC approved methodologies to ensure the Specified Acceptable Fuel Design Limits (SAFDLs), of which DNBR is one, are not violated. The current DNBR setpoint continues to ensure automatic protective action is initiated to prevent exceeding the proposed DNBR safety limit.

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

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

   Response: No.

   The proposed change does not result in any plant modifications or change in the way the plant is designed to function. The proposed change is not associated with any accident precursor or initiator. The proposed change supports the loading and use of Next Generation Fuel (NGF) at ANO-2 as previously approved by the NRC.

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

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

   Response: No.

   The use the NRC-approved NGF WSSV-T correlation with the ABB-NV correlation to establish a new bounding DNBR safety limit of 1.23, preserves the DNBR margin of safety at a 95/95 level. The Core Protection Calculator (CPC) DNBR power adjustment addressable constant BERR1 is calculated based on the WSSV-T and ABB-NV CHF correlations such that a CPC trip at a DNBR of 1.25 using the CE-1 CHF correlation assures that the bounding DNBR safety limit of 1.23 for the WSSV-T and ABB-NV CHF correlations will not be violated during normal operation and AOOs to at least a 95/95 probability / confidence level.

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

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

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

In letter dated June 19, 1981, the NRC approved ANO-2 proposed changes in the CHF correlation from the W-3 correlation to the CE-1 correlation and the associated thermal hydraulic methodology beginning in Cycle 2. Both the safety analysis and the CPC / COLSS algorithms were changed to the CE-1 correlation at that time.

Reference 7.8 documents the NRC's approval of the use of the WSSV CHF correlation and the associated thermal hydraulic computer code (VIPRE) with the ABB-NV CHF correlation.

7.0 REFERENCES

7.1 WCAP-16523-P-A, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," August 2007

7.2 WCAP-16500-P-A, "CE 16 x 16 Next Generation Fuel Core Reference Report," Revision 0, August 2007

7.3 WCAP-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel," Revision 000, May 2000

7.4 CENPD-161-P-A, "TORC Code – A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986

7.5 CENPD-162-P-A, "C-E Critical Heat Flux," September 1976; Supplement 1-A, February 1977, and CENPD-207-P-A, "C-E Critical Heat Flux Part 2 Nonuniform Axial Power Distribution," December 1984

7.6 CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties", Revision 01-P-A, May 1988

7.7 WCAP-16500-P, Supplement 1-P, Revision 1-P, "Application of CE Setpoint Methodology for CE 16 x 16 Next Generation Fuel (NGF)," October 2008

7.8 NRC Letter to Mr. J. A. Stall, Senior Vice President, Nuclear and Chief Nuclear Officer, Florida Power and Light Company, "St. Lucie Plant, Unit No. 2 – Issuance of Amendment Regarding Change in Reload Methodology and Increase in Steam Generator Tube Plugging Limit (TAC NO. MC1566)," dated January 31, 2005

7.9 CEN-139(A)-P, "Statistical Combination of Uncertainties – Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One Unit 2," November 1980

Attachment 2

2CAN050901

Proposed Technical Specification Changes (mark-up)

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained ≥ 1.23~~1.25~~.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor core has decreased to less than 1.23~~1.25~~, be in HOT STANDBY within 1 hour.

PEAK FUEL CENTERLINE TEMPERATURE

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-275-P, Revision 1-P-A and CENPD-382-P-A).

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak fuel centerline temperature has equaled or exceeded 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-275-P, Revision 1-P-A and CENPD-382-P-A), be in HOT STANDBY within 1 hour.

Attachment 3

2CAN050901

Proposed Technical Specification Bases Changes (mark-up)
For Information

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to ~~1.25~~1.23 for the ~~CE-1~~WSSV-T and ABB-NV correlations and is established as a Safety Limit. This value is based on a statistical combination of uncertainties. It includes uncertainties in the CHF correlation, allowances for rod bow and hot channel factors (related to fuel manufacturing variations) and allowances for other hot channel calculative uncertainties (CEN-356(V)-P-A, “Modified Statistical Combination of Uncertainties,” Revision 01-P-A, May 1988).

Second, operation with a peak linear heat rate ≤ 21 kw/ft setpoint will ensure that the peak fuel centerline temperature safety limit protects fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Limiting Safety System Setting. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

TS 2.1.1.2 establishes a peak fuel centerline temperature of 5080°F with adjustments for burnup and burnable poison. An adjustment for burnup of 58°F per 10,000 MWD/MTU has been established in NRC approved Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992. Adjustments for burnable poisons are established based on NRC approved Topical Reports CENPD-275-P, Revision 1-P-A, “CE Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers”, May 1988 and CENPD-382-P-A, “Methodology for Core Designs Containing Erbium Burnable Absorbers”, August 1993.

DNBR – Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor. At this pressure a DNBR – Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (ΔT) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.23~~1.25~~ such that the decrease in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit.

Since the CPC thermal hydraulic algorithm retains the CE-1 CHF correlation, the DNBR – Low trip setpoint and Allowable Value remain set at 1.25. The CPC DNBR power adjustment addressable constant BERR1 is calculated based on the WSSV-T CHF correlation such that a CPC trip at a DNBR of 1.25 using the CE-1 CHF correlation assures that the DNBR safety limit of 1.23 for the WSSV-T CHF correlation will not be violated during normal operation and anticipated operational occurrences to at least a 95/95 probability/confidence level.

CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modeling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.