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SUBJECT: Forwards TS,supporting operation of plant at increased core rated thermal power of 3588 MWt.

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AMERICAN ELECTRIC POWER

July 11, 1996

AEP:NRC:1223

Docket Nos. 50-315
50-316

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
LICENSE NOS. DPR-58 AND DPR-74
PROPOSED LICENSE AND TECHNICAL SPECIFICATION CHANGES
SUPPORTED BY ANALYSES TO INCREASE UNIT 2 RATED THERMAL POWER
AND
CERTAIN PROPOSED CHANGES FOR UNIT 1
SUPPORTED BY RELATED ANALYSES

This letter and its attachments constitute an application for amendment of the Donald C. Cook Nuclear Plant unit 2 facility operating license, for amendment of the Donald C. Cook Nuclear Plant unit 2 technical specifications (T/S), and for some related changes to the unit 1 technical specifications. Changes are proposed primarily to support operation of Cook Nuclear Plant unit 2 at an increased core rated thermal power of 3588 MWt. In addition, analyses and evaluations have been performed to support increased operating margins for unit 2. Changes to the T/Ss based on those analyses and evaluations are proposed. Some of the proposed changes are proposed for both unit 1 and unit 2.

The small break loss of coolant accident analysis, which is submitted with this letter, was performed using new Westinghouse Electric Corporation models. These models employ new methods for modeling safety injection to the broken loop and an improved steam condensation model. These models were submitted for NRC review by Westinghouse Electric Corporation under a letter dated December 14, 1994, identified as NTD-NRC-94-4278, to the Document Control Desk from N. J. Liparulo.

To be implemented, this submittal requires the approval of previous submittals. The first of these is identified as AEP:NRC:1207. It proposes "Technical Specification Changes Supported by Analyses to Increase Unit 1 Steam Generator Tube

A001
1/1

Plugging Limit and Certain Proposed Changes for Unit 2 Supported by Analyses." The submittal is dated May 26, 1995, from E. E. Fitzpatrick to the USNRC Document Control Desk. AEP:NRC:1207 includes the most recent steam line mass and energy release to containment analysis which bounds both units at a core power of 3588 MWt. That analysis, in part, provides support for the proposal to uprate unit 2 to a core power of 3588 MWt.

A description of the proposed changes and an analysis concerning significant hazards consideration pursuant to 10 CFR 50.92 is contained in Attachment 1. Attachment 2 contains the proposed, revised T/S pages. Attachment 3 contains the existing T/S pages marked to reflect the proposed changes. Attachment 4 is a summary description of the proposed T/S changes. It contains a brief summary of each proposed change and directs the reviewer to the supporting documentation. Attachment 5 is a discussion of earlier related submittals. The analyses described in the earlier submittals support future operation at the proposed, increased rated thermal power. These analyses, together with evaluations and analyses described in the attachments to this submittal, provide the necessary support for the proposed increase in rated thermal power. Attachment 5 also addresses previously submitted analyses for unit 1 that support the proposed changes to both units. Attachment 6 is a description of analyses performed by Westinghouse Electric Corporation to support the power uprate of Cook Nuclear Plant unit 2. Attachment 7 describes the effect of the proposed uprated power on balance of plant systems and the result of miscellaneous safety evaluations.

Plant radiation protection features are designed to limit the radiation exposure to plant personnel and the general public to 10CFR20 limits under normal conditions. While certain isotopes are present in greater concentration in the fuel gap due to the uprated power level, the actual increases in occupational dose are expected to be minimal. The fuel has been designed to operate at the higher power level without any damage, which would negate any increases in radioactivity trapped within an intact fuel rod. Also, our technical specifications limit the concentration of radioactivity within the reactor coolant system (see T/S 3.4.8). Nevertheless, accident offsite doses have been recalculated based on the uprated source term and other analysis assumptions used in the Uprating Program. In some cases, the resulting thyroid offsite dose consequences increase slightly above the values presently in the UFSAR. The new calculated whole body doses are bounded by the UFSAR values. For both types of calculations, the changes in the offsite radiation dose for each accident are not significant and are within the acceptance criteria as defined in 10CFR100.

Because systems and procedures controlling normal radioactive releases are based on limiting plant effluents to a small fraction of regulatory limits, the proposed uprating of unit 2 will not exceed 10CFR20 or 10CFR100 limits. Based on this information, there will be no significant increase in the types or amounts of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure.

Some of the changes proposed for unit 2 are also proposed for unit 1. The first proposes two footnotes that require the as left magnitude of the pressurizer safety valve tolerance be 1%. The second affects the Technical Specification Bases for containment internal pressure and air temperature. The final change affecting both units involves a clarification of the required volume for the condensate storage tank.

Our final review of two issues remains incomplete at this time. These issues are (1) the impact of power uprate on blowdown forces on ducting and cable trays in the containment and (2) the residual heat removal cooldown capability. We anticipate the final reviews will show that these issues can be safely addressed and will not adversely impact operation and licensing of unit 2 at an uprated core power level of 3588 MWt. Upon resolution of these issues, we will notify the NRC staff of the results.

Approval of the changes in this submittal is needed by August 1, 1997, to support unit 2, cycle 12 operation.

We believe the proposed T/S changes will not result in (1) a significant change in the types of effluents or a significant increase in the amount of effluent that might be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

These proposed T/S changes have been reviewed and approved by the Plant Nuclear Safety Review Committee and by the Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to the Michigan Public Service Commission and the Michigan Department of Public Health.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 11th DAY OF July 1996

Rita W Hill
Notary Public

My Commission Expires: 6-28-99

Attachments

cc: A. A. Blind
H. J. Miller
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

bc: S. J. Brewer/D. H. Malin/M. S. Ackerman
J. A. Kobyra/K. R. Baker/D. R. Hafer
J. B. Kingseed/V. D. Vanderburg/S. L. Colvis - w/attachments
D. F. Powell/S. K. Farlow/D. P. Schmader
B. B. Bradley - w/attachments (except attachment 6)
J. B. Shinnock
J. S. Wiebe
J. B. Hickman, NRC - Washington, D.C. - w/attachments
M. E. Eberhardt - w/attachments (except attachment 6)
M. E. Barfelz - w/attachments (except attachment 6)
G. P. Arent - w/attachments
PRONET - w/attachment
DC-N-6015.1

ATTACHMENT 1 TO AEP:NRC:1223

REASONS AND 10 CFR 50.92 ANALYSIS FOR
CHANGES TO THE
COOK NUCLEAR PLANT UNIT NO. 2
LICENSE AND TECHNICAL SPECIFICATIONS
AND
COOK NUCLEAR PLANT UNIT NO. 1
TECHNICAL SPECIFICATIONS

INTRODUCTION

The primary purpose of this submittal is to request approval to operate Cook Nuclear Plant unit 2 at an uprated core rated thermal power of 3588 MWt. The analyses needed to support this request include reanalysis, evaluation, or review of the events discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) and the evaluation of the capability of various systems and components. As discussed in Attachment 5, Discussion of Previous Related Submissions, previously submitted analyses for Cook Nuclear Plant unit 2 were, for the most part, performed at a rated thermal power of 3588 MWt. This was done to position unit 2 for operation at an uprated power. As discussed in Attachment 6, WCAP 14489, the previously analyzed events were reviewed to ensure that no evaluations were performed or any other event had occurred that would invalidate the analyzed core power. Events previously analyzed at a core power lower than 3588 MWt were reanalyzed or power sensitivity cases run to support the proposed uprated power. Analyses in this category are LOCA containment integrity analysis, large break LOCA with residual heat removal (RHR) crossties closed, and small break LOCA with high head safety injection (HHSI) crossties closed. The effect of operation at the uprated core power on NSSS systems and components is also addressed in Attachment 6 and the effect on balance of plant systems is addressed in Attachment 7. Miscellaneous safety evaluations have been included in Attachment 7.

Because the work needed to support the proposed increase in core power involved reanalysis or review of the events discussed in Chapter 14 of the UFSAR, analyses and evaluations were performed so that additional operating margin could be achieved in some areas. This submittal contains proposals based on these analyses and evaluations.

Furthermore, the currently approved unit 2 technical specifications (T/Ss) are based on analyses for a mixed core of Westinghouse Vantage 5 and Advanced Nuclear Fuel. Because the Cook Nuclear Plant unit 2 core now consists totally of Westinghouse Vantage 5 fuel, the penalties associated with the mixed core analysis are no longer appropriate. Therefore, a number of changes related to the completion of the transition to a full Vantage 5 core are proposed in this submission.

A few of the proposed changes to the T/Ss are applicable to both units. These changes relate to the LOCA containment integrity analysis that bounds both units and the correction of an omission in the proposed unit 1 T/Ss for relaxed pressurizer safety valve liftpoint tolerance. The unit 1 submission was made in AEP:NRC:1207, dated May 26, 1995. The corresponding proposal for unit 2 is included in the group of changes proposed to increase

unit 2 operating margin. It is a part of this submission. Also, the description of the required volume of the condensate storage tank has been changed from "contained" to "useable".

Finally, an administrative change is proposed. In Table 2.2-1, the design flow has been redefined as a fraction of the reactor coolant system total flow rate. The administrative change is discussed in more detail under the heading for Group 6.

The proposed changes are discussed in greater detail below and in Attachment 4, Summary Description of Proposed Unit 2 Power Uprate Technical Specifications. Attachment 4 is provided to assist the reviewer. It contains a brief summary of each change and directs the reviewer to the supporting documentation. Attachment 2 contains the proposed T/S changes. Attachment 3 contains the current T/S pages marked to reflect the proposed changes.

The summary in Attachment 4 provides a brief description of each proposed change and a cross reference to specific analyses where appropriate. The change in core power depends on the fact that all analyses needed to support the uprated core power have been completed at the uprated power; therefore, the references provided in Attachment 4 are general in nature.

DESCRIPTION OF CHANGES

The proposed changes are discussed in related groups.

Group 1: Changes Directly Related to Increased Rated Thermal Power

The Group I proposed changes are found in the operating license and the T/Ss listed below:

Operating License

Section 2.C(1) currently states, "Indiana and Michigan Electric Company is authorized to operate the facility at steady state reactor core power levels not in excess of 3411 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this license."

If the proposal to increase core rated thermal power is approved, the statement needs to be changed to, "Indiana and Michigan Electric Company is authorized to operate the facility at steady state reactor core power levels not in excess of 3588 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this license."

Technical Specifications

Increase rated thermal power
1.3

Reduce the applicability of the heatup and cooldown curves
Figure 3.4-2
Figure 3.4-3
B 3/4.4.9

Lower the maximum allowable power range neutron high flux setpoint with inoperable steam line safety valves

Table 3.7-1

The first group of these changes is directly related to the proposed increase in core rated thermal power.

The analyses that support the proposed uprating of Cook Nuclear Plant unit 2 have been performed over a period of years in several contexts. Including the new analyses described in Attachment 6 (WCAP 14489) and the evaluations described in Attachment 7 (Balance of Plant Evaluations and Miscellaneous Safety Evaluations), all the necessary analyses and evaluations have been completed to support an uprate of unit 2 to a core power of 3588 MWt. Except for the steam mass and energy release to containment submitted with submittal AEP:NRC:1207 (reference 30 of Attachment 5), the spent fuel pool thermal hydraulic analyses submitted with submittals AEP:NRC:1202 and AEP:NRC:1202A (as identified in the cover letter), and Attachments 6 and 7 of this submittal, all the analyses have been previously submitted and reviewed. A brief history of the development of the analyses supporting the uprated power is provided in Attachment 5. It summarizes the previous analyses that provide part of the support for uprated power and their associated submittals.

Attachment 6, WCAP 14489, describes the most recent analyses and sensitivity studies. New analyses have been performed to replace or supplement those analyses formerly performed at the currently approved maximum power level. The new analyses yielded acceptable results at the proposed uprated core power as described in Attachment 6. The Westinghouse model and the plant input assumptions were reviewed for the new long term containment analysis.

The revised input assumption with the greatest impact on the result was a newly revised structural heat sink model. The heat sink model was completely revised to reconstruct its basis. The analysis performed after this review was satisfactory.

The large break LOCA (LBLOCA) reanalysis with RHR cross ties closed was also satisfactory using the current model. As described in Section 3.1.1.3 of WCAP 14489, the new reanalysis incorporates model changes that resulted from the resolution of issues identified in 10CFR50.46 reports and in Westinghouse reports to the NRC. These model changes were a significant benefit to a Cook Nuclear Plant unit 2 specific analysis. The small break LOCA (SBLOCA) reanalysis also was satisfactory. As indicated in the cover letter of this submittal, the SBLOCA reanalysis was performed using the improved steam condensation model that results in a significant benefit to a Cook Nuclear Plant unit 2 specific analysis. This new analysis supports our proposal, described in Group 2, to delete the requirement in Emergency Core Cooling System Technical Specification that power be reduced when the HHSI crossties are closed.

Attachment 6 also summarizes analyses and evaluations previously performed by Westinghouse Electric Corporation to support the uprated core power for unit 2. Section 2.0 of WCAP 14489 references the earlier work. The evaluations described in WCAP 14489 are based on these earlier analyses. The earlier analyses are described in Rerating Program WCAP's 11902 and 11902 Supplement 1, references 3 and 10 of Attachment 5, and in the Vantage 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2, Revision 1, March 1990 (RTSR), reference 11 of Attachment 5. The Steam Generator Tube Plugging Program steamline mass and energy release (SGTP SM&E) to containment analysis is described in WCAP 14285, reference 29 of Attachment 5. WCAP 11902 and its supplement are referred to as the "Rerating Program" in WCAP 14489. The reload transition safety report is referred to as "RTSR" in WCAP 14489. The increase in the permitted level of steam generator tube plugging program is referred to as "SGTP Program" in WCAP 14489.

Attachment 6, together with earlier work referenced in Attachments 5 and 6, and Attachment 7 support the uprated core power for unit 2.

Group 2: Change to Remove Power Restriction for High Head Safety Injection Cross Ties Closed Operation

This group of proposed changes is found in the following T/Ss:

Delete power reduction requirement when the high head safety injection cross ties are closed
3.5.2
B 3/4.5.2 and B 3/4.5.3

The second group of changes consists of a single change to T/S 3.5.2, Emergency Core Cooling Systems. The currently approved Limiting Condition for Operation (LCO) requires that "all safety

injection cross-tie valves (be) open." If a high head safety injection cross-tie valve is closed, the ACTION statements must be entered and the core power reduced to 3250 MWt. The proposed change deletes these requirements.

The currently approved restriction in power with safety injection cross-ties closed is the result of the SBLOCA analysis performed in support of relaxing the main steam safety valve (MSSV) setpoint tolerance. The MSSV submittal and the associated SER are references 26 and 28 of Attachment 5, respectively.

The approved Westinghouse SBLOCA model at the time of the MSSV submittal required a power reduction to get an acceptable result. For this power uprating program, an improved SBLOCA model incorporating the COSI condensation model was used. The results were acceptable at the proposed uprated core power of 3588 MWt. The MSSV analysis and the new analysis with the COSI condensation model are discussed in more detail in Sections 3.1.2.3 and 3.1.2.4 of WCAP 14489, Attachment 6. The cover letter to this submittal addresses the fact that the SBLOCA analysis was performed using the new, improved model and provides a reference to the Westinghouse submittal for the new model.

Group 3: Changes Proposed to Increase Unit 2 Operating Margin

This group of proposed changes are found in the following T/S:

Revise Safety Limits and OPΔT/OTΔT Reactor Trip Setpoint
- Figure 2.1-1
- Table 2.2-1
- B 2.2.1 Overpower Delta T

Increase Unit 1 Pressurizer Safety Valve Tolerance
- 3.4.2
- 3.4.3

The third group of changes results from analyses and evaluations designed to increase operating margin. Because most of the events described in Chapter 14 of the UFSAR had to be reanalyzed, evaluated, or reviewed to ensure currency in order to support the increase in core rated thermal power, the effort to increase some margins was performed at the same time.

The first proposed changes in this group are changes to the overtemperature delta T (OTΔT) and overpower delta T (OPΔT) reactor trip setpoints. The new setpoints are based on core thermal safety limits for an all Vantage 5 core at 3588 MWt. The safety limits are those which were calculated for an all Vantage 5 core at the time of transition from Advanced Nuclear Fuel to Westinghouse fuel. The proposed safety limits could have been included with the changes either in this group or in Group 4 below. They are included here in Group 3 because of their relationship to the proposed OTΔT and OPΔT reactor trip setpoints.

As a result of temperature streaming in the reactor coolant hot legs, there is an inaccuracy in the measurement of T_{hot} in the resistance temperature detector (RTD) bypass lines. This streaming is a function of the core power distribution. Drift in the Delta T measurements at full power as a function of burnup results from this phenomenon. When the deviations exceed the instrument allowances for hot leg streaming, it is necessary to recalibrate the OTΔT and OPΔT system. The OTΔT and OPΔT reactor trip functions provide primary protection against fuel centerline melting, among other concerns (e.g., DNB and hot-leg boiling). Revised OTΔT and OPΔT reactor trip setpoints for the increased reactor core power of 3588 MWt were calculated to accommodate an increase in the allowance between the safety analysis limits and the technical specification setpoints.

The changes proposed in this submittal are based upon analyses performed by both us and our contractor, Westinghouse Electric Corporation. Westinghouse performed calculations to ensure that the safety analysis values for the OTΔT and OPΔT setpoints provide the necessary protection with respect to fuel centerline melting, the applicable core thermal limits, and that acceptable results are obtained for the affected transients. We performed the calculations to ensure adequate margin exists between the safety analysis values and the T/S nominal values of the OTΔT and OPΔT reactor trip setpoints. The associated allowable values proposed for notes 3 and 4 of T/S Table 2.2-1 in Attachments 2 and 3 are based on our calculations.

The pressurizer safety valve liftpoint tolerance was increased to ±3%. As indicated in Section 3.3.2.2 of Attachment 6, the appropriate events have been shown to support this increased tolerance.

The analyses and evaluations described in WCAP 14489, Attachment 6, support the proposed T/S changes to increase operating margin described above.

Group 4: Changes Related to Transition Core or Transition to Temperature Window/Dual Pressure Technical Specifications

This group of proposed changes is found in the following T/Ss:

Increase DNB Temperature Limit, Include Limits for Both Analyzed Pressures, Delete Low Temperature Limit
3.2.5
B 3/4.2.5

Reduce Setpoint and Allowable Value for SI on Low Pressurizer Pressure
Table 3.3-4

Reduce Setpoint and Allowable Value for SI on Low Steamline Pressure
Table 3.3-4

Reduce Setpoint and Allowable Value for Steamline Isolation on Low Steamline Pressure
Table 3.3-4

Include Pressure Criteria for Both Analyzed, Nominal Pressures
4.4.6.2.1

Remove References to Advanced Nuclear Fuel
B 2.1.1
B 3/4.2.2 and B 3/4.2.3

The fourth group of proposed changes are changes that remove restrictions related to operation of Cook Nuclear Plant unit 2 with a mixed core of Westinghouse Vantage 5 fuel and Advanced Nuclear Fuel. The first full Vantage 5 core was cycle 10. The proposed changes are planned for implementation in cycle 12. The proposals expand the temperature window to that analyzed for a full Vantage 5 core, identify both analyzed pressures, change engineered safety features actuation setpoints, delete the low temperature limit from the DNB T/S, and delete references to Advanced Nuclear Fuel from the bases.

Except for the proposal to lower the safety injection actuation setpoint on low pressurizer pressure and the proposal to delete the low temperature limit from the DNB T/S, the underlying analyses for the proposed changes in this group have been reviewed and approved as a part of previous submittals. However, the steam mass and energy release (SM&E) to containment analysis that directly supports the setpoint for low steam pressure was recently reanalyzed to bound both units at the unit 2 uprated power. This was done to correct some inaccurate analysis assumptions. As indicated in the cover letter and Attachment 5, this new analysis was submitted as part of our proposal to increase the limit of plugged steam generator tubes for unit 1 (SGTP), reference 30 of Attachment 5.

The analyses included in the Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant unit 2, revision 1, March 1990 (RTSR) generally addressed two situations, a mixed core of Advanced Nuclear Fuel and Westinghouse Vantage 5 fuel and a core of all Vantage 5 fuel. Operation with an all Vantage 5 core supported an operating temperature window and the option of operating at two primary pressures. The analyses for the mixed core used the W-3 DNB correlation for the Advanced Nuclear Fuel. Use of the W-3 correlation was a significant DNB penalty. To obtain acceptable results for a mixed core, the high temperature side of the temperature window was restricted to a Tavg of 576.0°F and operation was permitted only at the high nominal pressure of 2250 psia. These limitations are documented in the RTSR, reference 11 of Attachment 5. Operation of unit 2 with all Westinghouse fuel was approved by reference 17 of Attachment 5. Changes to support operation in the full temperature window and at both operating pressures are proposed in this submittal.

The proposed T/S changes include a new upper limit on reactor coolant system Tavg. The proposed departure from nucleate boiling (DNB) upper temperature limit was calculated from the upper limit of the temperature window for a full Vantage 5 core at a core power of 3588 MWt. The DNB temperature limit is obtained by adding the controller allowance to the high nominal Tavg used in the analysis and then subtracting the readability allowance. The high nominal Tavg is 581.3°F and the controller allowance is 4.1°F. These values are found in Table 3.3-1 and Section 3.3.3.1 of WCAP 14489, respectively. The readability allowance, calculated by AEPSC, is 2.1°F. The resulting DNB temperature limit is 583.3°F.

The proposed changes include adding the DNB pressure limit for low pressure operation. The DNB pressure limit is obtained by subtracting the total pressure allowance used in the analysis from the nominal operating pressure used in the analysis and then adding the readability allowance. The nominal pressures and the total allowance are found in Section 3.3.1 and 3.3.3.1 of WCAP 14489, respectively. The readability allowance, calculated by AEPSC, is 18.9 psi. The pressure limit currently in the T/Ss for high pressure operation is conservatively higher than the calculated value of 2191 psig. The proposed limit of 2050 psig for low pressure operation is an addition. It is conservatively higher than the calculated value of 2041 psig. The proposed value for the low pressure limit is the same as the unit 1 limit.

Other limitations of the transition analysis affect the engineered safety features actuation setpoints. At the time of the fuel transition, the reanalysis of SM&E outside containment for rerating and reduced temperature/reduced pressure operation was not complete. The evaluation of the then applicable analysis assumed an NSSS power of 3425 MWt and a low steam line pressure setpoint no lower than 520 psig. This limitation is documented in RTSR, reference 11 of Attachment 5. The revised SM&E release analysis outside containment needed to support the reduction in safeguards actuation setpoint on low steam line pressure was submitted in support of our proposal to lower the boron concentration in the boron injection tank (BIT), reference 24 of Attachment 5. As discussed in Attachment 5, this proposal was approved by reference 25 of Attachment 5.

The new SM&E inside containment, which was submitted with the unit 1 increased SGTP Program, and the core response steam line and feedwater line breaks submitted with the RTSR also support the proposal to lower the engineered safety features setpoint on low secondary pressure and its associated allowable value. Evaluations of the core response analyses are discussed in Sections 3.3.4.6 and 3.3.4.7 of WCAP 14489. The RTSR, its associated submittal and approval are references 11, 13, and 17 of Attachment 5. The SGTP submittal was addressed in the cover letter to this submittal and is reference 30 to Attachment 5.

As part of the "Rerating Program", an evaluation of margin to safety injection on turbine/reactor trip transients was performed. For unit 2 operating at Tavg above 570°F and at the low nominal pressure, it was determined that it would be necessary to reduce the safety injection actuation setpoint on low pressurizer pressure. Since this change is associated with operation at the lower of the two analyzed primary pressures, the proposal to lower this setpoint and its associated allowable value was included in this group of changes. The evaluations that support lowering the safety injection actuation setpoint on low pressurizer pressure are documented in Sections 3.3.4.5 and 3.3.4.6 of WCAP 14489. Other analyses affected by safety injection on low pressurizer pressure assumed a setpoint sufficiently low to accommodate the lowered setpoint.

The proposed changes delete the low temperature limit that currently appears in the unit 2 DNB specification. This proposal is included with the transition group (Group 4) because the low temperature limit is related to the analyzed temperature window. The proposed change converts the DNB specification back to a purely DNB specification. This is consistent with both the new and old standard T/S, NUREG-1431, Rev. 1 and NUREG-0452, Rev 4, respectively. The proposal will also make the unit 1 and unit 2 T/Ss more nearly alike. The cycle specific neutronics design imposes temperature limits that are more restrictive than either

the DNB limit or the low temperature limit proposed for removal. Compliance with the cycle specific temperature limits is controlled administratively.

The other changes in this group are the removal of references to Advanced Nuclear Fuel. The references to Advanced Nuclear Fuel are not needed because unit 2 cores are now all Westinghouse Vantage 5 fuel.

Group 5: Changes Proposed for Both Units

This group of proposed changes is found in the following T/S's:

Add Footnote to Pressurizer Safety LCO Requiring 1% as Left Tolerance

3.4.2
3.4.3

Change Peak Containment Pressure to Reflect New Analysis, Both units
B 3/4.6.1.4
B 3/4 6.1.5

Change Required Condensate Storage Tank from Contained to Useable.
3.7.1.3
4.7.1.3.1
B 3/4.7.1.3

Group 5 proposes changes applicable to both units. These changes are in three categories. The first proposes a footnote requiring the as left tolerance of the pressurizer safety valve tolerance be 1%. This requirement is consistent with a similar requirement approved for the main steam safety valves. It is being submitted for both units because it was inadvertently omitted in our submittal AEP:NRC:1207, dated May 26, 1995, which included the analytical justification for an increase in pressurizer safety valve setpoint tolerance for unit 1.

The second change in this group affects the T/S bases. The peak pressure of the long term containment integrity analysis is now being reported as being below the limit of 12 psig instead of reporting the specific value calculated in the analysis. The new analysis reported in WCAP 14489, Attachment 6, bounds both units at a core power of 3588 MWt. The proposed change to the T/S bases bounds the value calculated in the new analysis.

The third proposal changes the contained volume of the condensate storage tank to useable volume in Technical Specifications 3.7.1.3 and 4.7.1.3.1. The proposed Basis Section 3/4.7.1.3 is

also revised to address why useable volume is appropriate. Due to the fact that the zero of the level instrumentation is located at the centerline of the discharge pipe, above the level for required NPSH, all the indicated volume is useable. Therefore, there is no need to address any allowance for water not useable.

Group 6: Administrative Change

This group of proposed changes is found in the following T/S's:

Redefine Design Flow to be 1/4 of Reactor Coolant System Total Flow Rate
Table 2.2-1

The final group consists of an administrative change. The proposal changes the design flow footnote in Table 2.2-1 to a reference to Reactor Coolant System Total Flow Rate of Specification 3.2.5. Design flow is 1/4 of Reactor Coolant System Total Flow Rate. This change ensures that there is only one place in the T/S's to change this parameter. The proposed change has no substantive impact.

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

10 CFR 50.92 specifies that the holder of an operating license or construction permit of a nuclear power facility participate in determining whether a change to the T/S's or license involves a significant hazards consideration. Prior to implementation of a change to the T/S's or license, the Nuclear Regulatory Commission must review and make a final determination, pursuant to the procedures in 10 CFR 50.91, that a proposed amendment involves no significant hazards considerations. To satisfactorily complete the review, the proposed amendment must not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated,

2. create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. involve a significant reduction in a margin of safety.

For the purpose of performing a significant hazards consideration analysis, the six groups of T/S and operating license changes discussed under Description of Changes can be reduced to three groups. In evaluating significant hazards, those changes supported by analyses, essentially all of the first five groups

of proposed T/S and operating license, will be considered together. The removal of the low temperature limit from the DNB T/S and the administrative change will each be considered separately.

DETERMINATION OF NO SIGNIFICANT HAZARDS FOR CHANGES BASED ON ANALYSES AND EVALUATIONS [Groups 1, 2, 3, 4 (except deletion of DNB low temperature limit), and 5 (except condensate storage tank useable volume)]

Criterion 1

Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The analyses performed to support the first five groups of proposed changes demonstrate that plant equipment will operate acceptably at the uprated conditions and applicable acceptance criteria are met. The proposed T/S and operating license changes do not involve postulated initiators for analyzed events; therefore, the probability of accidents can not be affected. The analyses and evaluations performed all met applicable acceptance criteria; therefore, the consequences of accidents previously evaluated are unaffected.

Criterion 2

Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The first group of proposed changes increases the core power at which unit 2 may be operated. Operation at the proposed new power has been analyzed. The second group of changes proposes to remove the power restriction when the HHSI cross ties are closed. HHSI is an accident mitigator. The proposed changes in this group are based on a new analysis using an improved model. The analyses performed to support the third, fourth, and fifth groups of proposed changes address increases in operating margin for accident mitigators. No new accident is involved in this proposal.

Criterion 3

Do the proposed changes involve a significant reduction in a margin of safety?

No. The margin of safety is provided for the primary pressure boundary and other components in part by applicable design codes. The margin of safety for the various accidents and transients is maintained by the analysis acceptance criteria. Because the components remain in compliance with the codes and standards in effect when Cook Nuclear Plant was licensed and applicable acceptance criteria are met, the margin of safety is not reduced by the proposals in this unit 2 uprate program.

DETERMINATION OF NO SIGNIFICANT HAZARDS FOR DELETION OF THE LOW TEMPERATURE LIMIT FROM THE DNB TECHNICAL SPECIFICATION (Within Group 4)

Criterion 1

Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposal to delete the low temperature limit from the DNB T/S change does not involve a physical change to the plant. The procedures and administrative controls for the plant described above will either remain in place or be replaced by controls of comparable effectiveness. Therefore, the proposed T/S change will not result in a significant increase in the probability or consequences of any accident previously analyzed.

Criterion 2

Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change to the DNB T/S does not involve a physical change to the plant. The procedures and administrative controls for the plant described above will either remain in place or be replaced by controls of comparable effectiveness. Therefore, the proposed change will not create the possibility of a new or different accident from any previously evaluated.

Criterion 3

Do the proposed changes involve a significant reduction in a margin of safety?

No. The proposed change to the DNB T/S does not involve a physical change to the plant. The procedures and administrative controls for the plant described above will either remain in place or be replaced by controls of comparable effectiveness. Therefore, the proposed T/S change will not involve a significant reduction in any margin of safety.

DETERMINATION OF NO SIGNIFICANT HAZARDS FOR ADMINISTRATIVE CHANGES [Group 5 (Condensate Storage Tank Useable Volume only) and Group 6]

Criterion 1

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

No. The proposed changes involve referencing another T/S rather than incorporating a specific value into a second specification and a change in terminology reflecting the existing instrument configuration. These changes are for convenience and have no substantive impact. These proposals have no impact on probability. The proposed changes also have no impact on the consequences of an accident because they have no substantive impact on plant operation or operating limits.

Criterion 2

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

No. Nothing is changed with regard to accident initiators. There is no substantive change; therefore, the proposed changes can have no impact on accident initiators.

Criterion 3

Does the proposal involve a significant reduction in a margin of safety?

No. The proposal does not change any requirements; therefore, there is no change in the margin of safety.

CONCLUSION

It is concluded that operation of Cook Nuclear Plant units 1 and 2, with the changes proposed above, does not involve any significant hazards consideration as defined in 10 CFR 50.92.

ATTACHMENT 2 TO AEP:NRC:1223

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