UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE STEAMLINE BREAK ANALYSIS

AND AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated August 15, 1988, Indiana Michigan Power Company (the licensee) proposed an amendment to the Technical Specifications appended to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit 2. The proposed revisions are supported by the steamline break analysis performed by Advanced Nuclear Fuels (ANF). The analysis was performed using the methodology described in Reference 1 in response to comments by the NRC staff regarding previous ANF analyses. The analysis, described in report XN-NF-87-31(P), was transmitted to the NRC by the May 29, 1987 ANF letter and placed on the D. C. Cook docket by the licensee via a letter dated June 15, 1987.

The proposed Technical Specification changes are:

1. Increase in required shutdown margin from 1.6% delta-k/k to 2.0% delta-k/k to reflect the ANF analysis assumption.

2. Inclusion of a time response testing requirement of $\leq$ 10 seconds for the high steam flow/low-low T_{ave} steamline isolation function to reflect the ANF analysis assumption.

3. Reduction in allowable end of life moderator temperature coefficient from -3.9×10^{-4} delta-k/k/°F to -3.5×10^{-4} delta-k/k/°F to reflect the ANF analysis assumption.

4. Change in end of life moderator temperature coefficient surveillance due to the ANF analysis and desire to improve the surveillance basis.

5. Change in expression of high steam flow/low-low T_{ave} setpoint from percent full steam flow to lbs/hr to improve Unit 1/Unit 2 similarities.

2.0 EVALUATION

2.1 Steamline Break Analysis

The steamline break analysis performed by ANF is described in report XN-NF-87-31(P) (Ref. 2). The analysis utilizes the methodology which is

described in report XN-NF-84-93(P), Steamline Break Methodology for PWRs (Ref. 1). This methodology was approved by the NRC staff in Reference 3. The ANF analyses utilize RELAP5, XCOBRA-IIIC, and XTG computer codes to predict the plant and core response to a steamline break. The analysis assumptions, plant modeling, and computer code interfaces are such that a conservative estimate of the plant and core response is predicted.

The analysis for D. C. Cook Unit 2 provided in Reference 2 included four possible transient scenarios. The scenarios included initiation of the steamline break transient from Hot Zero Power (HZP) or Hot Full Power (HFP) and with or without the availability of offsite power. The fuel response for the four scenarios was evaluated against minimum departure from nucleate boiling ratio (MDNBR) and centerline melt linear heat generation rate (LHGR) criteria. No fuel failures were predicted to occur based on either MDNBR or LHGR limits.

The HZP with loss of offsite power scenario was determined to be limiting with respect to MDNBR. The initiation from HZP results in a higher return to power. The coastdown of the reactor coolant pumps caused by the loss of offsite power reduces the return to power but the combination of power and reduced flow resulted in the lowest MDNBR for the cases analyzed. Reactor trip and safety injection are predicted to be actuated by the differential pressure between steam lines function. Main steam isolation is assumed to occur due to the high steam flow/low-low T_{ave} actuation signal. Delivery of borated water from the Emergency Core Cooling System (ECCS) is limited due to the assumed failure of one of two charging pumps and conservative modeling such as a stagnant reactor vessel upper head. Cooldown of the primary coolant and resultant power increase are maximized by the break flow model, feedwater and auxiliary feedwater modeling, stuck rod assumption, and other aspects of the methodology and plant specific analysis. The predicted MDNBR is above the Modified Barnett Correlation safety limit of 1.135, and therefore no fuel failure is expected to occur related to the DNBR critera.

The HZP with offsite power available scenario was determined to be limiting with respect to the centerline melt LHGR criteria. The initiation from HZP and continued operation of the reactor coolant pumps result in the maximum predicted power increase. Safety system initiations and conservative assumptions are similar to the HZP with loss of offsite power scenario. The analysis results predict a peak LHGR less than the 21 Kw/ft limit just prior to the borated ECCS water from one of two charging pumps reaching the core.

The ANF analysis of a steamline break for D. C. Cook Unit 2 has been found to be conservative. The methodology includes a conservative modeling of the break flow (Ransom-Trapp steam only) and adequately represents the asymmetric response of the reactor coolant system and reactor core. The upper head of the reactor vessel was modeled to maximize the possibility of flashing and related delay of ECCS delivery. Plant specific assumptions regarding feedwater and auxiliary feedwater delivery, failure of one of two charging pumps, stuck rod location, and initiation setpoints and delays associated with safety systems were found to be conservative with respect to either the current plant requirements or those Technical Specification amendments requested by the licensee.

2.2 Technical Specifications

The Technical Specification amendments proposed by the licensee are:

Shutdown Margin

The proposed Technical Specification revises the required shutdown margin for Modes 1 through 4 from a value of 1.6% delta-k/k to 2.0% delta-k/k. This change is in the more restrictive direction and is required to reflect the ANF assumption in the steamline break analysis. The proposed change has been reviewed and found to adequately incorporate the analysis assumption and is therefore acceptable.

Time Response Testing

The proposed Technical Specification adds an Engineered Safety Features Response Time requirement (Table 3.3.-5) for the steamline isolation function on high steam flow with coincident low-low T_{ave}. This addition is required due to reliance upon this function and specific actuation signal in the ANF analysis. The requirement of $\leq$ 10 seconds agrees with the ANF assumption and has been determined to be acceptable.

Moderator Temperature Coefficient (MTC)

The proposed revision in the allowable end of life MTC from -3.9×10^{-4} to -3.5×10^{-4} delta-k/k/°F is more restrictive and reflects the ANF assumption in the steamline break analysis. The proposed change has been reviewed and found to adequately incorporate the analysis assumptions and is therefore acceptable.

The proposed revision in the end of life MTC surveillance changes the criteria to which the measured MTC is compared upon reaching an equilibrium boron concentration of 300 ppm. The existing Technical Specification requires the measured MTC at 300 ppm to be less negative than -3.0×10^{-4} delta-k/k/°F. This value corresponds to the end of life analysis assumption of -3.9×10^{-4} delta-k/k/°F and ensures conservatism upon reaching the end of life core conditions. The required change in the end of life MTC limit to -3.5×10^{-4} delta-k/k/°F also required an inspection of the surveillance requirement. The licensee decided to use an extrapolation to end of life conditions and compare directly with the -3.5×10^{-4} delta-k/k/°F criteria instead of defining a value for 300 ppm which would ensure meeting the end of life criteria. The increased surveillance requirements (measure every 14 EFPD) associated with measuring an MTC more negative than the limit is maintained. The extrapolation to end of life for comparison to the actual MTC analysis assumption would also make the Unit 2 Technical Specifications more similar to those of Unit 1.

The proposed revision has been reviewed and found to adequately ensure the end of life MTC analysis assumption remains bounding for actual D. C. Cook Unit 2 cores. Maintaining the increased surveillance requirements ensures that required actions are taken prior to exceeding the end of life limit.

High Steam Flow/Low-Low Tave Setpoint

The proposed revision changes the expression of the steamline isolation setpoint from percent full steam flow to lbs/hr. The change results in minimal if any actual change in the differential pressure setpoints but allows the terminology for the setpoint to be similar for Units 1 and 2. The proposed change reflects the ANF assumption in the steamline break analysis and has been found acceptable.

2.3 Conclusion

The ANF analysis of a steamline break for D. C. Cook Unit 2 has been reviewed and found acceptable. The methodology, modeling, and selection of plant-specific assumptions has allowed the identification of limiting scenarios and conservative predictions which show that applicable acceptance criteria are satisfied. The proposed Technical Specification revisions have been found to be either required by the steamline break analysis or adequately supported by the analysis, and as such, have been determined to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment and Finding of No Significant Impact has been issued for this amendment (54 FR 6976, February 15 1989).

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. XN-NF-84-93(P), "Steamline Break Methodology for PWRs," Exxon Nuclear Company, November 1984.

2. XN-NF-87-31(P), "Steam Line Break Analysis for D. C. Cook Unit 2, "Advanced Nuclear Fuels Corporation, May 1987.

3. Letter, A. Thadani (NRC) to R. A. Copeland (ANF), subject: Acceptance for Referencing of Licensing Topical Reports, ANF-84-93(P) and ANF-84-93(P), Supplement 1, "Steamline Break Methodology for PWRs," December 28, 1988.

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