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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 70 TO LICENSE NO. DPR-46

## NEBRASKA PUBLIC POWER DISTRICT

## DOCKET NO. 50-298

## COOPER NUCLEAR STATION

### 1.0 INTRODUCTION

Nebraska Public Power District (the licensee) requested amendments to the Technical Specifications for the Cooper Nuclear Station (CNS) by letter dated March 5, 1981. The amendments are associated with core Reload Number 6, Cycle 7 operation.

### 2.0 CORE RELOAD NUMBER 6

### 2.1 Introduction

By letter(1) dated March 5, 1981, the Nebraska Public Power District (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-46 for Cooper Nuclear Station (CNS). The proposed changes relate to the refueling of CNS. This reload involves the replacement of 32 exposed 7x7 fuel assemblies and 80 exposed 8x8 assemblies with an equivalent number of fresh, two water rod, P8x8R fuel assemblies designed and fabricated by the General Electric Company (GE). In support of this reload application for CNS, the licensee has submitted a supplemental reload licensing document(2) prepared by GE and proposed plant Technical Specification changes.(3)

The descriptions of the nuclear and mechanical design of the fresh P8x8R fuel assemblies the exposed 8x8R fuel assemblies and the exposed standard 8x8 fuel assemblies, which were used in connection with prior CNS reloads, are contained in GE's generic licensing topical report(4) for BWR reloads. Reference 4 contains a complete set of references to other GE topical reports which describe GE's BWR reload methodologies for the nuclear, mechanical, thermal-hydraulic, transient and accident analysis calculations. Information addressing the applicability of these methods to reload cores containing a mixture of 7x7, 8x8, 8x8R and P8x8R fuel is also contained in Reference 4. Portions of the plant-specific data, such as operating conditions and design parameters used in transient and accident calculations, have also been included in the topical report.

8106050177

Our safety evaluations[6,7] of GE's generic reload licensing topical report and supplement concluded that the nuclear and mechanical design of the 8x8R and P8x8R fuel and GE's analytical methods for the nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing different fuel types are acceptable.

As part of our evaluations[6] of Reference 4, we found the cycle-independent input data for the reload transient and accident analyses for CNS to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 2, which follows the format and content of Appendix A of Reference 4. Finally, the licensee has changed the initial core pressure used in the transient analyses from 1045 psia to 1035 psia, to reflect actual plant operating data.

As a result of the staff's generic evaluations[6,7] of a substantial number of safety considerations relating to the use of P8x8R reload fuel in mixed core loadings with 7x7, 8x8 and 8x8R fuel, only a limited number of additional review items are included in this evaluation of Cycle 7 of CNS. These items include the plant and cycle-specific input data and safety analysis results presented in Reference 3, those items identified in our evaluation[6] as requiring special consideration during reload reviews, and the proposed Technical Specification changes.[3]

## 2.2 Evaluation

### 2.2.1 Nuclear Characteristics

Reload 6 consists of 112 new P8x8R fuel bundles with bundle average enrichments of 2.83 and 2.65 wt% U-235. The remainder of the 548 fuel assembly reconstituted core will consist of irradiated 7x7, 8x8, 8x8R and P8x8R fuel assemblies exposed during earlier cycles. The assumed cycle exposure has increased from 17,110 MWd/t for Reload 5 to 17,441 MWd/t for Reload 6. The reference core loading for Reload 6 is shown in Figure 1 of Reference 2 will result in quarter core symmetry, which is consistent with previous cycles.

The reload application follows the procedure described in Reference 4. We have reviewed this application and the consequent Technical Specification changes. The transient analysis input parameters provided in Section 6 of Reference 2 are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients and the required operating limit values for MCPR are given for each fuel type. The revised MCPR limits are required by the reload and they are acceptable.

### 2.2.2 Thermal Hydraulics

#### Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 4, for BWR cores which reload with GE's P8x8R fuel, the allowable minimum critical power ratio (MCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this MCPR safety limit during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) to be used for Cycle 7 is unchanged from the SLMCPR previously approved for Cycle 6. The basis for this safety limit is addressed in Reference 4, while our generic approval of the limit is given in References 6 and 7.

#### Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed 7x7, 8x8 and 8x8R fuels and for the fresh P8x8R fuel. Addition of the largest reductions in critical power ratio ($\Delta$ CPR) to the safety limit MCPR establishes the operating limits for each fuel type.

The transients evaluated were the limiting pressure and power increase transient (in this case, the load rejection transient without turbine bypass to the main condenser), the limiting coolant temperature decrease transient (loss of feedwater heater), the feedwater controller failure transient, the control rod withdrawal error transient and the fuel loading error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 2 were assumed.

The nonpressurization transients were analyzed using the methods described in Reference 4. As per a letter to all BWR Licensees from the NRC[8] all operating BWRs reload submittals with General Electric analyses received after February 1, 1981 are requested to have the limiting transients (overpressurization) recalculated with the ODYN code. The proposed technical specification changes for Cycle 7 will incorporate the requirements of ODYN (Option B) and new MCPR limits associated with these new analyses.

The calculated system responses and $\Delta$CPRs for the above listed operational transients and conditions have been analyzed by the licensee. Results for 100% power/100% flow core conditions were as follows:

TABLE 1

| Transient | Exposure | $\phi$ (% NBR) | Q/A (% NBR) | $P_{SL}$ (psig) | $P_v$ (psig) | ΔCPR 7x7 | ΔCPR 8x8 | ΔCPR P8x8R & 8x8R |
|---|---|---|---|---|---|---|---|---|
| Load Rejection w/o Bypass | BOC-EOC | 501.5 | 122.3 | 1179 | 1213 | 0.14 | 0.19 | 0.21 |
| Loss of 100°F Heater Feedwater | -- | 123 | 121.7 | 1022 | 1069 | 0.12 | 0.14 | 0.14 |
| Feedwater Controller Failure | -- | 314.4 | 119.1 | 1135 | 1172 | 0.09 | 0.13 | 0.14 |
| Rod Withdrawal Error @10% RBM Set Point | | | | | | | 0.16 | 0.08 |

The operating MCPR is dependent on the cycle average scram time ($\tau$ave) and the adjusted analysis scram time ($\tau$B). If $\tau$B > $\tau$ave, the limiting event is a rotated fuel bundle (fuel loading error). The operating MCPR(s) are obtained from the fuel loading error analysis. However, if $\tau$ave >$\tau$B the limiting event is load rejection without bypass and the ODYN option B method is used to determine the limiting MCPR(s).

Based on the most limiting transient the licensee has proposed the following operating limit MCPRs (OLMCPR's).

TABLE 2

| | 7x7 | 8x8 | 8x8R | P8x8R |
|---|---|---|---|---|
| Fuel Loading Error | 1.23 | 1.24 | 1.24 | 1.24 |
| ODYN OLMCPR's* | T/S Fig 3.11-2a | T/S Fig 3.11-2b | T/S Fig 3.11-2c | T/S Fig 3.11-2d |

*Determined from ODYN Option B

Since the higher OLMCPR obtained from the analyses will preclude violation of the safety limit MCPR of 1.07 in the event of any anticipated operational occurrence, we find these limits to be acceptable.

2.2.3 Overpressure Analyses

For Cycle 7 the licensee has reanalyzed the limiting pressurization event to demonstrate the ASME boiler and pressure vessel code requirements are met. The overpressure analysis for the MSIV closure with high flux scram has been performed in accordance with the requirements of Reference 7.

The sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig. The reanalysis shows that the peak pressure at the bottom of the reactor vessel does not exceed 1290 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. Therefore, the limiting overpressure as analyzed by the licensee is acceptable.

2.2.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 2) shows that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line intersection (which is the least stable physical attainable point of operation) are within the stability limit. Decay ratio for Reload 6 was 0.80 as compared to 0.78 for Reload 5. Since operation in the natural circulation mode will be prohibited by the Technical Specifications, there will be added margin to the stability limit and this is acceptable to the staff. This predicted decay ratio is below the 1.0 ultimate performance limit decay ratio proposed by General Electric.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating with a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

2.3 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance, the licensee will perform a series of physics startup tests, which was described in Reference 9. This test program was submitted previously in connection with the Cycle 5 reload. Our Cycle 5 review found this program to be acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee for staff review following completion of the Cycle 7 tests.

2.4 Technical Specifications

The licensee has submitted proposed changes to the Cooper Technical Specification (3). The effects of these changes are to change the MCPR limits to make them consistent with the values presented in Table 2 of this evaluation. Based on the analysis results, these changes to the MCPR limits in the Technical Specifications are found to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 22, 1981

REFERENCES

1. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito) dated March 5, 1981.

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 6," Y1003J01A17, January 1981.

3. "Proposed Changes to the Cooper Nuclear Station Technical Specification" appearing as an Enclosure to the NPPD letter to USNRC dated March 5, 1981.

4. "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-1 - July 1979.

5. General Electric BWR Thermal Analysis Basis (GETAB): "Data Correlation and Design Application," General Electric Company, BWR Systems Department, November 1973 (NEDO-10958).

6. USNRC letter (D. Eisenhut) to General Electric (R. Gridley) dated May 12, 1978.

7. USNRC letter (T. Ippolito) to General Electric (R. Gridley) dated April 16, 1979.

8. NRC letter (D. Eisenhut) to all BWR licensees, dated November 4, 1980.

9. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito) dated April 16, 1979.