April 13, 2000

Mr. J. H. Swailes
Vice President of Nuclear Energy
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - CORRECTION TO SAFETY EVALUATION WITH RESPECT TO AMENDMENT ON AVERAGE POWER RANGE MONITOR NEUTRON FLUX-HIGH (FLOW-BIASED) ALLOWABLE VALUE (TAC NO. MA7705)

Dear Mr. Swailes:

On April 11, 2000, the Nuclear Regulatory Commission issued Amendment No. 184 to Facility Operating License No. DPR-046 for the Cooper Nuclear Station. The amendment revised the Technical Specifications with respect to the Average Power Range Monitor Neutron Flux-High (Flow-Biased) allowable value. After issuance, a minor error was discovered in the safety evaluation. The containment response analysis evaluation should reflect that the maximum peak drywell temperature (301.4 ^{0}F) at the point of maximum vessel subcooling occurs at the conditions of 102 percent power/75 percent flow (Case 3 in Nebraska Public Power District's (NPPD's) February 15, 2000, submittal) not at 62 percent power/34 percent flow (Case 4 in NPPD's February 15, 2000, submittal). The enclosed page reflects the correct information and should replace page 4 of the April 11, 2000, safety evaluation.

Please contact me if you have questions concerning this matter. I may be reached at (301) 415-3053 or ljb@nrc.gov.

Sincerely,

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure: As stated

cc w/encl: See next page

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Cooper Nuclear Station

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the MELLL condition was an estimated increase of 50 ^{0}F from the GE8x8NB fuel design in the Cycle 20 reload core at rated core flow. The analysis of the GE14 fuel design at the ICF condition resulted in no change in PCT compared to rated core flow. The evaluation was conducted using the approved GESTAR-II methodology SAFER/GESTAR-LOCA. The GE14 Licensing Basis PCT of 1760 ^{0}F was calculated at the MELLL point which is well below the 10 CFR 50.46, 2200 ^{0}F limit. Therefore, the analysis and results are acceptable.

3.4 Containment Response

Bounding short-term containment response analyses of the design-basis LOCA event were performed to demonstrate that operation in the MELLL and ICF domains will not result in exceeding containment design limit. The CNS updated safety analysis report short-term containment evaluation is dependent on reactor initial pressure and temperature that change with MELLL and ICF operation. The long-term heatup of the suppression pool following a LOCA is governed by the decay heat which depends on the reactor rated power level which remains unchanged with either MELLL or ICF operation.

For the short-term containment response of the design-basis LOCA, the maximum peak drywell pressure is 54.4 psig at 102 percent power and 75 percent core flow, which is below the design value of 56 psig. The maximum peak wetwell pressure is 24.0 psig, which is well below the design value of 56 psig. Therefore, this value is acceptable to the staff.

The maximum peak drywell temperature is 301.4 ^{0}F at the point of maximum vessel subcooling (102 percent power/75 percent flow), which is below the maximum allowable drywell airspace temperature acceptance criterion. In its letter dated February 15, 2000, NPPD stated that this acceptance criterion was 309 ^{0}F, which is contained in CNS TS Bases Section B 3.6.1.5. In its letter dated April 8, 2000, NPPD stated that this acceptance criterion may require revision and is being evaluated within NPPD's Corrective Action Program. However, NPPD stated that this does not impact the acceptability of the calculated peak drywell airspace temperature because the revised acceptance criteria will be at least equal to 301.4 ^{0}F. Based on the calculated maximum peak drywell temperature being less than or equal to the maximum allowable drywell airspace temperature acceptance criterion, the staff finds this acceptable.

3.5 Reactor Vessel Internals Pressure Differences and Structural Integrity

Operation in either the MELLL or ICF domain affects the pressure differences across reactor internal components. Operation in the ICF region (up to 105 percent rated flow) results in higher initial flow velocities and causes increased reactor internal pressure differences (RIPD) across the internal components for Normal, transient (Upset), Emergency, and accident (Faulted) conditions. The impact of MELLL operation on RIPD is bounded by ICF operation.

Analyses of Normal operating conditions were performed with the steady-state thermal-hydraulic model at 100 percent power/105 percent flow. The inputs used for this analysis are consistent with the original CNS RIPD with the assumption of a full core of the limiting fuel (GE14) for pressure drop consideration.

For Upset conditions, the steady-state (Normal condition) values are conservatively adjusted to obtain the limiting AOO RIPDs. However, the initial steady-state pressure differences at the low flow conditions (MELLL and ELLL) would be smaller than for ICF at the same power level because of the lower initial flow velocity. Consequently, it is bounding to apply Upset conditions adjustment factors to the conservative ICF steady-state results.