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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 62 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 letters dated February 8. 1980; March 17, 1980 as revised April 18, 1980, February 26, 1980; and October 9, 1978. The amendments are associated with core Reload Number 5, the suppression system downcomers, diesel generator periodic tests, and certain administrative improvements respectively.

2.0 CORE RELOAD NUMBER 5

2.1 Introduction

By letter(1) dated February 8, 1980, 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 fifth refueling of CNS. This reload involves the replacement of 72 exposed 7x7 fuel assemblies and 40 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)

This reload (Reload 5) is the first for CNS to utilize GE's new prepressurized 8x8R fuel design. Previously for Reload 4, 100 unpressurized retrofit 8x8R assemblies were loaded into the core. In addition, numerous other BWRs have already refueled once with the new GE prepressurized fuel design while lead prepressurized retrofit test assemblies, previously loaded into another operating reactor, have performed satisfactorily for at least two cycles.

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

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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.

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. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was provided in the staff's evaluation(8) of the information contained in Reference 9.

As part of our evaluation(6) of Reference 4, we found the cycle-independent if the nuclear and mechanical design of the standard 8x8 fuel was provided in the staff's evaluation(8) of the information contained in Reference 9.

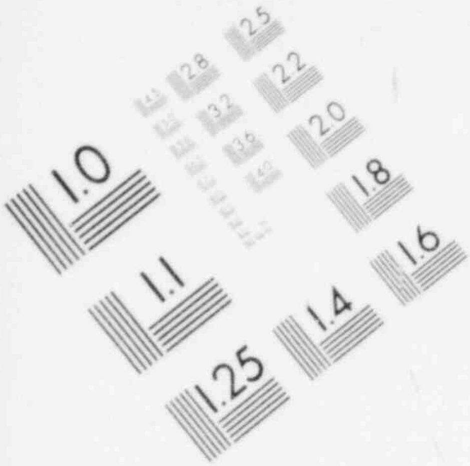
As part of our evaluation(6) of Reference 4, we found the cycle-independent input data for the reload transient and accident analyses for CNS-1 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 2, which follows the format and content of Appendix 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 6 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

For Cycle 6, up to 112 fresh P8x8R fuel bundles with bundle average enrichments of 2.83 wt/% and 2.65 wt/% U-235 will be loaded into the core, replacing an equal number of exposed 7x7 and 8x8 assemblies. The remainder of the 548 fuel assembly reconstituted core will consist of irradiated 7x7, 8x8 and 8x8R fuel assemblies exposed during earlier cycles. Thus, about 20 percent of the fuel bundles are being replaced for this reload. The reference core loading for Cycle 6, which is shown in Figure 1 of Reference 2, will result in quarter core symmetry, which is consistent with previous cycles.



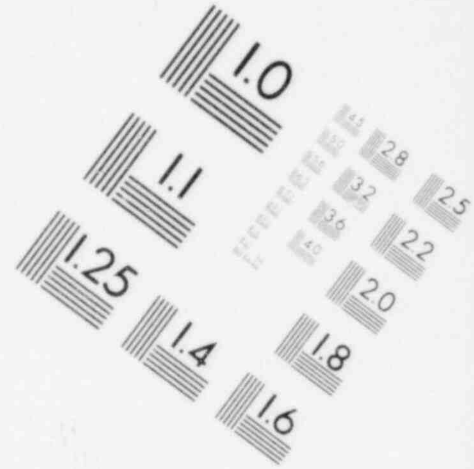
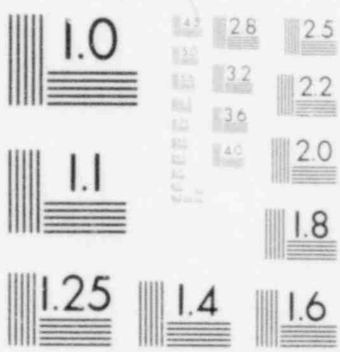


IMAGE EVALUATION
TEST TARGET (MT-3)



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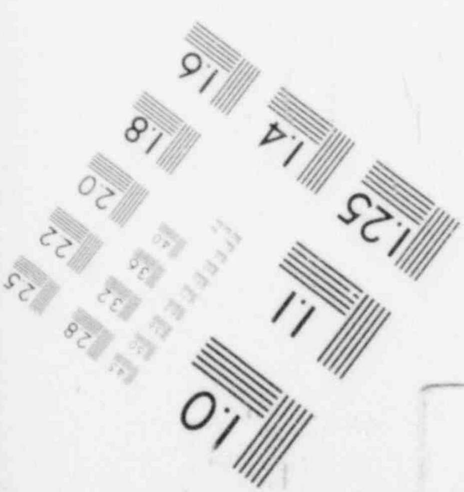
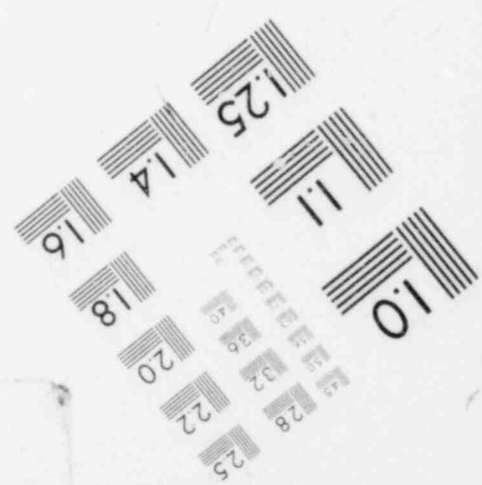
MICROCOPY RESOLUTION TEST CHART





The information provided in Section 6 of Reference 2 indicates that the fuel temperature and void dependent characteristics of the refueled core are not significantly different from previous cycles of CNS-1. Additionally, scram effectiveness, as shown in Figure 2 of Reference 2, is also similar to earlier cycles. The 1.3%Δk/k calculated design shutdown margin for the reconstituted core meets the Technical Specification requirement that the core be subcritical by at least 0.25%Δk/k in the most reactive operating state when the single most reactive control rod is fully withdrawn and all other rods are fully inserted. Finally, Reference 2 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 4.3%Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

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 6 is unchanged from the SLMCPR previously approved for Cycle 5. 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 to the safety limit MCPR establishes the operating limits for each fuel type.

Abnormal Operational Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 4. Our acceptance of the cycle-independent values appears in References 6 and 7. Additionally, our evaluation of the transient analysis methods, together with a description and summary of the outstanding issues associated with these methods, appears in Reference 6. Supplementary cycle-dependent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of Reference 2. Our evaluation(6) of the methods used to develop these supplementary input values has also been completed.

At the time we completed our evaluation of the generic methods, the acceptability of the GEXL critical power correlation(5), for use in connection with the retrofit fuel design, had not been adequately documented by GE. The staff found, however, that the then available 8x8R (P8x8R) critical power test data was sufficient to support the acceptability of GE's 8x8R and P8x8R fuel designs for BWR core reloads for one operating cycle. Accordingly, we stated(6) that future BWR core reload applications involving retrofit 8x8 fuel for a second operating cycle would have to include additional information which adequately justified the correlation for application to 8x8R fuel operating beyond one cycle. Subsequent to our approval of Reference 6, GE provided a report(10) to the staff on this matter, together with additional information(11) intended to justify the adequacy of the GEXL correlation for application to the retrofit fuel over its design lifetime.

Reference 10 provides the results of full scale critical power tests performed on 8x8R fuel bundles. The tests, which included both transient and steady-state simulations, followed the same approved procedures(5) used for the standard 8x8 (single water rod) and 7x7 (all fueled rods) fuel designs. The analysis of a total of 577 steady-state data points was performed using methods also previously approved by the staff. The data, involving nine test assemblies, spanning a range of local power peaking and flow conditions, showed according to GE, that the GEXL correlation was applicable to the retrofit fuel if adjustment were made to the additive constants used in the formulation of the rod-by-rod R-factors. The local power peaking dependent R-factors, used by the GEXL correlation to evaluate 8x8R bundle critical power, are based on the new additive constants shown in Figure 3-1 of Reference 12, which were also used for the CNS-1, Cycle 5, 8x8R critical bundle power predictions. Using these new additive constants, GE performed a data analysis to assess the accuracy and precision of the GEXL correlation. The results of this analysis showed that the correlation fit provides for a mean predicted-to-measured critical power ratio of 0.9879 with a standard deviation of 0.0234.

When viewed over the range of its applicability (which is the same as the standard 8x8 fuel), the GEXL correlation is therefore somewhat conservatively biased while the statistical variation between the predicted and measured critical power is somewhat less than that associated with the standard 8x8 assembly[5], i.e., 2.34% vs 2.8%. Thus, when viewed over its range of applicability, the 8x8R GEXL correlation (with new additive constants) has somewhat better precision in predicting 8x8R critical bundle powers than the 7x7 and 8x8 GEXL formulations are for predicting 7x7 and 8x8 critical bundle powers respectively. Furthermore, from these results it may also be concluded that the 3.6% standard deviation and best estimate assumption of the GEXL correlation (which were actually used in the GETAB statistical analysis to derive the 1.07 safety limit MCPR) bound the statistical characteristics associated with the subject 8x8 GEXL correlation.

The additional information furnished by GE is also intended to be applicable to all BWR cores which contain 8x8R fuel. Accordingly, this information is also currently being generically reviewed by the staff. Although our evaluation is not yet complete, based on our review to date, we believe that for the range of testing, the 8x8R GEXL correlation has an acceptability and applicability which is equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. From our review of the subject data to date, we have also observed that for those critical power test conditions specifically representative of second cycle fuel operating at a normal operating thremal-hydraulic state point, the correlation is somewhat nonconservative in its predictions. This observation focuses in on a correlation behavioral concern not explicitly addressed in the overall GETAB methods approved[5] for the 7x7 and 8x8 fuel types.

Again, this subject is being generically reviewed by the staff. However, although this review is not yet complete, we believe that for Cycle 6 of CNS, there will continue to be sufficient conservatism implicit in the generic determination of the 1.07 safety limit MCPR to offset a possible nonconservatism associated with this concern. That is, specifically, the generic GETAB statistical analysis assumed a 3.6% correlation uncertainty while GE's analysis of the 8x8R test data results in a 2.34% standard deviation. Additionally, the generic evaluation considered an all 8x8R equilibrium core whereas the Cycle 6 CNS core involves a substantial number of 7x7 and 8x8 fuel assemblies together with the fresh and exposed 8x8R fuel assemblies in a non-equilibrium condition. In view of these conservatisms (which are representative of a typical non-equilibrium 8x8R reload core) we believe that the overall thermal-hydraulic (GETAB) methods are adequate for establishing conservative MCPR operating limits for Cycle 6 of CNS. However, as 8x8R (P8x8R) equilibrium conditions are approached, this conservatism will diminish. In order that this conservatism not be substantially eroded with future reload cycles, this issue should be addressed for the next reload of CNS.

Abnormal Operational Transient Analysis Results

The transient events analyzed for this reload were of the following types: pressurization load rejection without bypass and feedwater controller failure), feedwater temperature reduction (loss of 100°F feedwater heating) and local reactivity insertion (control rod withdrawal error).

The licensee reports that the most limiting event in the above categories for the exposed 7x7 assemblies, the exposed 8x8R assemblies and the reload P8x8R asemblies is the load rejection without bypass. For the 7x7 fuel this transient results in a CPR reduction of 0.12, while for the 8x8R and P8x8R fuel it results in a 0.18 change. The most limiting transient for the exposed 8x8 assemblies is the control rod withdrawal error, which results in a 0.18 change in critical power ratio with a revised Average Power Range Monitor rod block setpoint of 107%. Addition of these ΔCPRs to the 1.07 SLMCPR establishes fuel-type dependent operating limit MCPRs (i.e. 1.19 for the 7x7 fuel, 1.25 for the 8x8 fuel and 1.25 for the 8x8R/P8x8R fuel) sufficient to assure that the SLMCPR will not be violated during Cycle 6 for any of the aforementioned events.

The licensee has also considered the effects of the most severe fuel loading errors on bundle ΔCPR as well as the MCPR requirements as related to the analysis of the Loss of Coolant Accident. The results and requirements of these analyses are discussed in Section 2.2.3 herein.

Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawal error and fuel loading error events were reanalyzed by the licensee to determine the maximum transient linear heat generation rates (LHGRs). The results for CNS Cycle 6 show that the fuel type and exposure-dependent safety limit LHGRs, shown in Table 2-3 of Reference 4, will not be violated should these events occur. Thus, fuel failure due to excessive cladding strain will be precluded should either of these events occur. We find these results, which adequately account for the effects of fuel densification power spiking, to be acceptable.

2.2.3 Accident Analysis

ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading... "the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

For Cycle 6, the licensee has reevaluated the adequacy of CNS ECCS performance in connection with the new prepressurized fuel designs (type P8DRB265L and P8DRB83), using methods previously approved by the staff. The results of these plant specific analyses are given in Reference 2.

We have reviewed the information submitted by the licensee and conclude that CNS will be in conformance with all of the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when operated in accordance with the MAPLHGR versus Average Planar Exposure values appearing in Section 14 of Reference 2.

Finally, the ECCS evaluation model (SCAT) fuel assembly transient heat flux calculation assumes an initial steady state minimum critical power ratio of 1.20 for each fuel type. The fact that the Cycle 6 reanalysis of abnormal operational transients (see Section 2.2.2.2) would allow a lower minimum operating limit CPR of 1.19 for the 7x7 fuel types in no way alters the assumption used in the 7x7 MAPLHGR analysis. Accordingly in order to assure that the 1.20 MCPR value assumed in the ECCS analyses will not be violated, the licensee has proposed a MCPR limit of 1.20 for the 7x7 fuel during Cycle 6 rather than the 1.19 determined solely by transient requirements. This is acceptable.

Control Rod Drop Accident

For Cycle 5, the key plant-specific and cycle-specific nuclear characteristics for the worst case control rod drop accident (CRDA) occurring during both hot startup and cold start up conditions are conservatively bounded by the values used in bounding CRDA analysis given in Reference 4. The bounding analyses, which includes the adverse effects of fuel densification power spiking, shows that the peak fuel enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 6 of CNS, the peak enthalpy associated with with a CRDA from either the hot or cold startup conditions will also be within the 280 cal/gm design limit.

Fuel Loading Error

The licensee has considered the effect of postulated fuel loading errors on bundle CPR. An analysis of the most severe fuel loading errors was performed using GE's revised analysis methods which have previously been reviewed and approved by the staff. The results show that the worst possible fuel bundle misloadings will not cause a violation of the 1.07 safety limit MCPR assuming the proposed OLMCPRs. These results include the application of a 0.02 penalty factor applied to the CPR results of the misoriented fuel bundle analysis, as required by our approval of the revised methods. Thus, these operating limit MCPRs will effectively preclude DNB related fuel failures caused by either fuel cladding overheating or cladding oxidation, which might otherwise occur because of a fuel loading error. These results are acceptable to the staff.

2.2.4 Overpressure Analysis

For Cycle 6, the licensee has reanalyzed the limiting pressurization event to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met for CNS. The methods used for this analysis, when modified to account for one failed safety valve, have also been previously approved[6] by the staff. 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 1315 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is acceptable to the staff.

2.2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 4. The results show that the fuel type dependent channel hydrodynamic stability decay ratios and reactor core stability decay ratio at the least stable operating state (corresponding to the intersection of the natural circulation power curve and the 105% rod line) are 0.28 (8x8R/P8x8R), 0.37 (8x8), 0.22 (7x7) and 0.78 respectively. These predicted decay ratios are well below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

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.

Prior to Cycle 5 operation, the staff as an interim measure, added a requirement to the CNS Technical Specifications which restricted planned plant operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during Cycle 6. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of CNS during Cycle 5 to be acceptable.

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 6 tests.

2.4 Technical Specifications

The proposed Technical Specification changes(3) for Cycle 6 include: reviewed operating limit minimum critical power ratios (MCPRs) for each fuel type in the core, a new rod block monitor (RBM) setpoint and the addition of new MAPLHGR versus average planar exposure values for the two new prepressurized fuel types.

The licensee has proposed a single operating limit MCPR of 1.20 for the 7x7 fuel type and 1.25 for the 8x8, 8x8R and P8x8R fuel assembly types. Based on our evaluation appearing in Sections 2.2.2 and 2.3.1 herein, the staff finds these operating limit MCPRs to be consistent with and adequately supported by the Reload Safety analyses. The licensee has proposed to increase the flow biased RBM trip level setting from 105% to 107% at full flow. The change was proposed in order to allow increased control rod withdrawal maneuverability during power operation, while at the same time preventing the rod withdrawal error from becoming an overly limiting transient event for any fuel type. Since the revised setpoint is consistent with and adequately supported by the safety analysis, as evaluated in Section 2.2.2 herein, we find the proposed change acceptable. Finally, the licensee has proposed limits as shown in Figures 3.11-6, and 3.11-7 of Reference 3. As discussed in Section 2.3.1, these values are acceptable.

3.0 SUPPRESSION CHAMBER DOWNCOMERS

Introduction

By Reference 13, as amended by Reference 14, the licensee requested an amendment to the CNS Technical Specifications in conjunction with Mark I Containment Long-Term Program (LTP) modifications. The proposed amendment would reduce the maximum suppression chamber downcomers submergence to 3.0 feet 4 inches and would reduce the maximum differential pressure between the drywell and the suppression chamber from 1.47 to 1.0 psid. The licensee is shortening the length of downcomers as part of the LTP and, additionally, has determined that the existing Technical Specification limitation on drywell to suppression chamber differential pressure cannot be maintained with the shortened downcomers.

3.2 Evaluation

One method of suppression pool hydrodynamic load mitigation that the Mark I Owners Group has adopted for the LTP is reducing the initial submergence of the downcomer in the suppression pool to a minimum of three feet. By shortening the length of the downcomer, the pool volume (i.e., thermal capacity) of the original design would be maintained. This approach, however, raises concern regarding the increased potential for uncovering the downcomers and steam condensation capability, both of which could lead to torus overpressurization.

3.2.1 Seismic Slosh

The potential for downcomer uncovery is addressed in the assessment of seismic slosh. This assessment was performed at the most extreme conditions that could potentially lead to uncovering of the downcomers and was predicted on a minimum three-foot downcomer submergence.

Seismic motion induces suppression pool waves which can (1) impart an oscillatory pressure loading on the torus shell, and (2) potentially lead to uncovering the ends of the downcomers, which would result in steam bypass of the suppression pool and potential overpressurization of the torus, should the seismic event occur in conjunction with a Loss of Coolant Accident (LOCA). To assess these effects, the Mark I Owners Group undertook the development of an analytical model which would provide plant-specific seismic wave amplitudes and torus wall pressures. This model was based on 1/30-scale "shake test" data for a Mark I torus geometry (Reference 15).

Based on the results of plant-specific analyses, using the analytical model, the Mark I Owners Group concluded that (1) the seismic wave pressure loads on any Mark I torus are insignificant in comparison with the other suppression pool dynamic loads, and (2) the seismic wave amplitudes will not lead to uncovering the downcomers for any Mark I plant. This conclusion was based on the maximum calculated pressure loads and the minimum wave through depth relative to the downcomer exit.

We have reviewed comparisons of the analytical predictions with scaled-up test data, the small-scale test program, and the seismic spectrum envelope used in the plant-specific analyses. Based on this review, we conclude that the seismic slosh analytical predictions will provide reasonably conservative estimates of both the wall pressure loading and the wave amplitude, for the range of Mark I plant conditions.

Since the maximum local wall pressures were found to be less than 0.8 psi at a 95% upper confidence limit, the Mark I Owners Group has proposed that the seismic slosh loads may be neglected in the structural analysis. We agree that the seismic slosh loads are insignificant in comparison with the other suppression pool dynamic loads. On this basis, we conclude that neglecting seismic slosh loads for the plant-unique analyses is acceptable.

The results of the slosh wave amplitude predictions indicate that, within the local area of maximum amplitude and with maximum suppression pool drawdown (resulting from ECCS system flows), the slosh waves will not cause uncovering of the downcomers. We have reviewed the assumptions used in these analyses and conclude that they are sufficiently conservative. Based on the above discussion, we find the proposed change acceptable.

3.2.2 Condensation Capability

Condensation capability of the suppression pool is a function of the local pool temperature in the vicinity of the downcomer exit. Full Scale Test Facility (FSTF) test results (Reference 16) and foreign test data (Reference 11) have shown that thermal stratification occurs, and becomes more severe as the downcomer submergence is reduced. The most severe thermal stratification has been observed in low flow tests with a quiescent pool. However, in actual plant conditions, the Residual Heat Removal (RHR) system and Safety Relief Valve (SRV) discharge provide sufficient long-term pool mixing to minimize thermal stratification. Even with vertical thermal stratification, we have determined that the high energy reposition is accompanied by an increased flow and mixing, which prevent overpressurization of the torus. In addition, the analytical predictions of the torus pressure and bulk temperature response have been found to be conservative when compared with FSTF test data for plant simulated initial conditions. The local temperature variation in the pool which has been observed in the test data is not significant to the structure, and, therefore, need not be considered in the structural analysis.

Base. on this assessment, we conclude that a minimum initial downcomer submergence of three feet is acceptable, and there is sufficient conservatism in the containment response analysis techniques to accomodate the effects of thermal stratification.

3.2.3 Differential Pressure

The introduction of a positive pressure differential between the drywell and the suppression chamber air volume reduces the height of the water leg inside the downcomers. The reduced water leg permits the downcomers to clear earlier in the LOCA transient with the drywell consequently at a lower pressure. This effect reduces both the downward and upward pressure loads on the torus. The CNS plant-unique minimum differential pressure was reviewed and approved (Reference 18) by the staff as part of the Short-Term Program (STP).

The licensee has considered the effect of the shortened downcomers in conjunction with the reduced differential pressure and found that the torus support system and the torus support piping will continue to meet all of the Mark I Containment Short Term Program requirements following the modification (Reference 14).

Since the proposed modification retains the safety margin for torus pressure loads previously found acceptable for the STP (Reference 15), we conclude that, in the interim until the LTP is completed, the proposed modification is acceptable. Therefore, we find the proposed Technical Specifications acceptable.

4.0 DIESEL GENERATOR TESTING

4.1 Introduction

The present CNS Technical Specifications require that the emergency diesel generators be tripped, restarted, and reloaded after the sequential loading test that is performed each refueling outage. The licensee by Reference 19 requested the deletion of the requirement for restarting and loading after the sequential test.

4.2 Discussion

The existing Technical Specification 4.9.A.2.C presently reads as follows:

> "Once every 18 months, it will be demonstrated that there is no desirable interaction between the onsite power source (diesel generators) and the offsite power source (startup transformer) by simulating interruption and subsequent reconnection of onsite power sources to their respective buses."

This has been interpreted to require that the emergency diesel generators be tripped, restarted, and reloaded after the sequential loading test that is performed each refueling outage. The licensee's request for deletion is based on an attempt to minimize challenges to the Emergency Core Cooling System. The licensee has also stated that the test required by the subject specification is a test to verify the design of the electrical system control logic, rather than an operational performance test.

Preoper. al testing has been performed at Cooper Nuclear Station that verified the control logic design and it was verified that there was no undesirable interaction between onsite and offsite power sources. The licensee has also stated that since the control logic design has not been changed, it is unnecessary to continue the design testing as a surveillance requirement of the Technical Specifications.

4.3 Evaluation

The purpose of the periodic test that requires the diesel generators be tripped, restarted and reloaded was to verify that the onsite power system design includes the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. We agree with the licensee that this test does not demonstrate that there is not an undesirable interaction between onsite power sources and offsite power sources. The preoperational testing performance at Cooper Nuclear Station that verified the design of the electrical system control logic and determined that there was no undesirable interaction between onsite and offsite power sources is considered to be adequate since the design has not been changed. We also agree that it is unnecessary to continue this design testing as a surveillance requirement of the Technical Specifications.

The load shedding and load sequency capabilities of the onsite power sources are verified by the periodic diesel generator tests performed per the requirements of Cooper Nuclear Station Technical Specification 4.9.A.2.b. Manual interruption of the onsite source breaker and subsequent reconnection is a repeat of the above periodic diesel generator test and would increase challenges to the Emergency Core Cooling Systems connected to the buses fed by the diesel generators and may degrade the overall effectiveness of those systems. Based on this we agree that the proposed deletion of the subject Technical Specification will minimize challenge to the Emergency Core Cooling Systems.

Based on our review of the licensee's submittal, we find the proposed deletion of Technical Specification 4.9.A.2.c acceptable.

5.0 ADMINISTRATIVE IMPROVEMENTS

5.1 Introduction

By Reference 20, the licensee proposed Technical Specification changes removing surveillance requirements for a non-safety related switch, changing the frequency of emergency plan drills to meet current NRC requirements, and updating the Cooper Nuclear Station organization chart.

5.2 Evaluation

A temperature indicating switch has been deleted from Technical Specification Tables 3.2A and 4.2A which pertain to primary containment and reactor vessel isolation instrumentation. The switch in question isolates the filter demineralizer units upon high inlet temperature thereby preventing damage to the ion exchange resins. There is no safety isolation function credited to this switch. Deletion of the monthly testing requirement eliminates unnecessary thermal cycling and thus, reduces the possibility of accelerated corrosion. This change is acceptable.

The frequency of emergency plan drills required in Technical Specification 6.3.8 is reduced from twice per year to once per year. This frequency is consistent with Regulatory Guide 1.101, Revision 1 and is considered adequate.

The Cooper Nuclear Station Organization chart (Figure 6.1.2) has been modified to change the title of the Reliability Engineer on the Station Superintendent's staff to Technical Assistant, Figure 6.1.2 as well as Technical Specification 6.1.3F is also revised to increase the number of unlicensed operators during reactor operation from two to three. The additional operators are necessary to staff the fire brigade. Both of these organizational changes are acceptable.

6.0 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.

7.0 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 20, 1980

8.0 References

1. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito) dated February 8, 1980.

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 5," NEDO-24230, December 1979.

3. "Proposed Changes to the Cooper Nuclear Station Technical Specification" appearing as an Enclosure to the NPPD letter to USNRC dated February 8, 1980.

4. "Generic Reload Fuel Application," NEDE-24011-PA, August 1978.

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. "Status Report on the Licensing Topical Report 'General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel' NEDO-20360 Revision 1 and Supplement 1" by the Division of Technical Review, ONRR, USNRC, April 1975.

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

10. General Electric letter (R. Gridley) to USNRC (D. Eisenhut and D. Ross), dated October 5, 1978, transmitting "General Electric Information Report NEDE-24131, Basis for 8x8 Retrofit Fuel Thermal Analysis Application."

11. General Electric letter (R. Engle) to USNRC (D. Eisenhut and R. Tedesco), dated March 30, 1979.

12. USNRC letter (G. Lear) to Nebraska Public Power District, dated May 2, 1978.

13. Letter, J. Pilant (NPPD) to T. Ippolito (NRC) dated March 17, 1980.

14. Letter, J. Pilant (NPPD) to T. Ippolito (NRC) dated April 18. 1980.

15. S. M. Arian, "Mark I Containment Program Seismic Slosh Evaluation" GE Proprietary Report NEDE-023702-P, March 1978.

16. G. W. Fitzsimmons and others, "Mark I Containment Program Full Scale Test Program Final Report" GE Proprietary Report NEDE-2453q-P April, 1979.

17. K. W. Wong, "Mark I Containment Program Downcomer Reduced Submergence Functional Assessment Report" General Electric Proprietary Report NEDE-21885-P, June 1978.

18. NRC Report, "Mark I Containment Short-Term Program - Safety Evaluation Report," NUREG-0408, December 1977.

19. Letter, J. Pilant (NPPD) to T. Ippolito (NRC) dated February 26, 1980.

20. Letter, J. Pilant (NPPD) to T. Ippolito (NRC) dated October 9, 1978.