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SAFETY EVALUATION OF THE OFFICE OF NUCLEAR REACTOR REGULATION OF REVISION 1 TO DPC-NE-3005-P UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY DUKE ENERGY CORPORATION OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

1.0 INTRODUCTION

Duke Energy Corporation (DEC) first received approval from the NRC staff to perform transient analyses for core reloads at the Oconee Nuclear Station Units 1, 2, and 3 in 1981. This methodology has been updated and revised on several occasions. Topical Report DEC-NE-3005-P describes the methodology that is currently used for Oconee (Reference 1). The initial review by the staff of DEC-NE-3005-P is discussed in Reference 2. In Reference 2, the staff found that the DEC methodology, as documented in DPC-NE-3005-P, was acceptable, with the exception of the following four items:

1. A peak reactor coolant pressure acceptance criterion of 110 percent of the design pressure for the postulated locked-rotor accident should be included.

2. The proposed steam generator tube rupture methodology should be modified to assume no operator action at the beginning of the event and for consistency with the updated final safety analysis report (UFSAR), which credits the low-pressure reactor trip.

3. The licensee should modify the proposed methodology for large and small main steamline break (MSLB) events to assume failure of main feedwater to isolate. The Oconee plants are not equipped with safety-related main feedwater isolation capability.

4. Since the occurrence of a small steamline break is considered to be of moderate frequency, the acceptance criteria for events of this type should include the requirement of no fuel failure.

DEC made the required modifications, which were submitted for NRC staff review as Revision 1 to DPC-NE-3005-P (Reference 3).

2.0 EVALUATION

The NRC staff verified that the requirements from the previous staff review had indeed been incorporated into the topical report. The acceptance criteria for a postulated locked-rotor accident now include the requirement that the reactor system pressure remains below 110 percent of design pressure. The analysis of a steam generator tube rupture has been modified to assume automatic reactor trip 20 minutes after the steam generator tube rupture. This value is conservative in comparison with the analysis performed by DEC as documented in the UFSAR. In the UFSAR analysis, reactor trip was calculated to occur in 8 minutes if reactor operators take no remedial action. Actually operators are instructed to reduce reactor power at a rapid rate and to increase charging flow. The reduction of power and increase in charging may prevent a reactor trip. Preventing reactor trip would lessen the likelihood of a challenge to the main steam safety valves and would, thereby, reduce the offsite dose consequences.

The Oconee nuclear plants are not equipped with safety-related main feedwater isolation valves that would actuate following an MSLB accident. The NRC staff, therefore, required the licensee to perform analyses of an MSLB, assuming continued operation of the main feedwater system. The additional feedwater might cause recriticality in the core and a return of fission power generation even after reactor trip. The combination of low reactor system pressure and core power generation might cause the critical heat flux (CHF) to be exceeded thus leading to fuel cladding damage.

The licensee analyzed the reactor system aspects of MSLBs using the RETRAN-02 computer code. RETRAN-02 calculates, among other parameters, the break flow from the broken steamline, the heat transfer from the primary system through the steam generator tubes, the reactor power, and condition of the fluid entering the reactor core.

A break in a main steamline will cause the coolant loop with the break, the "affected" loop, to cool much more rapidly than the rest of the primary coolant system. The licensee calculates the effect of this asymmetric cooldown by dividing the core into two halves, each associated with one of the coolant loops. Mixing between the two loops at the core inlet plenum is input to RETRAN-02 by means of a mixing coefficient. The mixing coefficient was derived from a series of tests at Oconee Unit 1 in which a temperature differential was introduced between the reactor system cold legs while the reactor was at power. Reactor vessel temperatures at various locations were monitored.

The once through steam generators at Oconee do not contain steam separation equipment. A large steamline break would, therefore, be expected to result in a substantial amount of liquid being expelled through the broken steamline. The licensee uses modeling in the RETRAN-02 code, which inhibits liquid entrainment but, at the same time, maintains full steam generator heat transfer. With continued uncontrolled feedwater flow into the affected steam generator, no liquid entrainment is permitted until the steam generator is almost full. Then liquid is permitted to flow out of the broken steamline. These assumptions are designed to conservatively maximize steam generator heat transfer.

The analysis includes the reactivity and flux peaking effect of the maximum-worth control assembly remaining outside the core following a reactor trip. The amount of local flux peaking is calculated using the SIMULATE-3P computer code. SIMULATE-3P is a core neutronics code used to generate multi-dimensional core power distributions. The maximum-worth control assembly is conservatively assumed to be in the cooler half of the core, which will experience the greatest increase in local reactivity and power. A 10 percent reduction in the worth of the remaining control assemblies is also assumed.

The VIPRE-01 computer code is used to calculate the CHF within the core, including the hot channel. For steamline break analysis, the licensee developed VIPRE-01 input that models the entire core. The entire core was modeled so that flow redistribution from the hottest fuel bundle could be described. Simulation of the entire core with flow redistribution produces a more conservative result than modeling it as a single channel. The staff has previously approved use of RETRAN-02, SIMULATE-3P, and VIPRE–01 computer codes for use in Oconee licensing analyses (Reference 2).

Calculation of heat transfer between the primary system and the affected steam generator is significant in evaluating the cooling of the reactor system. The depressurization of the affected steam generator will cause an increase in boiling heat transfer between the tubing exterior and secondary coolant. The Thom Nucleate Boiling Correlation is used in RETRAN-02. The Chen Nucleate Boiling Correlation is programmed into the RELAP5 MOD3 code, which was developed by the staff. The licensee provided an evaluation of the accuracy of the Thom correlation in predicting nucleate boiling test data (Reference 4). The evaluation showed no significant difference between the predictions of the Thom and Chen correlations. The Thom correlation was shown to calculate slightly higher heat transfer in comparison to the test data which would be conservative for MSLB evaluations.

The core flood tank model in RETRAN-02 assumes the cover gas and liquid to be at the same temperature during the discharge of a flood tank. The NRC staff questioned the conservatism of this assumption since if the cover gas cooled faster than the liquid, a smaller flood tank discharge flow rate might be predicted and less boric acid would be introduced into the core to prevent core recriticality. The licensee stated that it assumes a conservatively low flood tank initial temperature, which compensates for the non-equilibrium effect. The licensee further states that the DEC RETRAN-02 core flood tank model was benchmarked against Oconee full-scale blowdown tests and that the model showed good agreement with the data.

The licensee performed sensitivity studies to evaluate the effects of single failures. Various combinations of reactor coolant pump operation, main feedwater flow control, and availability of offsite power were considered. The most severe single failure was found to be failure of one train of high-pressure injection (HPI). HPI adds boric acid to the reactor system and acts to dampen any return to power calculated to result from overcooling of the reactor system.

With offsite power available, the reactor coolant pumps, as well as the feedwater and condensate systems, could remain operable following an MSLB. Continued main feedwater flow to the affected steam generator would overcool the reactor system, increase core power generation, and reduce the reactor system pressure. Continued reactor coolant pump operation would also increase overcooling by facilitating heat removal from the reactor system. Reactor coolant pump operation would, however, act to maintain core heat transfer and the margin to CHF.

The licensee evaluated continued feedwater system operation with and without assumed operation of the integrated control system (ICS) to control steam generator level. The licensee calculated a return to criticality using both assumptions. The reactor coolant pumps were assumed to be operating, but the reactor coolant pumps in the unaffected loop were assumed to trip at 100 seconds into the event when voiding occurred in that loop. The RETRAN-02 coolant pump model became unstable under these conditions. The greatest return to power (13.09 percent) was calculated to occur if the ICS was assumed to remain operating, controlling steam generator level. The licensee believes that this is the result of increased boiling in the steam generator for the ICS operable case. For the ICS failure case, RETRAN-02 predicts that the steam generator will gradually fill with liquid after the initial blowdown and will experience less boiling.

The staff performed an audit calculation of this event using the RELAP5 MOD3 computer code. In the RELAP analysis, the ICS was assumed to fail so that main feedwater flow was uncontrolled. The affected steam generator filled and liquid flowed out the steamline. All the reactor coolant pumps were assumed to remain operating. The reactor coolant pump model in RELAP did not become unstable when voiding occurred in the unaffected reactor coolant loop. A very low value of decay heat was assumed to maximize overcooling. The RELAP code calculated slightly less reactor system cooling than did the licensee's analysis. RELAP also calculated lower reactor system pressures. The lower pressures calculated by RELAP resulted from not assuming the tripping of the reactor coolant pumps in the unaffected loop which, if tripped, would allow that loop to become stagnant. The lower reactor system pressure calculated by RELAP allowed additional boric acid to be injected by the HPI system and the core flood tanks. The core remained subcritical in the staff's calculation. Therefore, the licensee's analysis was conservative.

Reactor operators are trained to manually trip all reactor coolant pumps if the difference between the reactor system temperature and the saturation temperature (subcooling margin) becomes sufficiently small. This action is taken because for a certain range of small reactor coolant system break sizes, unacceptable results might be obtained if the reactor coolant pumps were not manually tripped early into the event. A large MSLB would resemble a small reactor system break in that the subcooling margin would also be lost and HPI would be actuated. Reactor operators might be expected to manually trip the reactor coolant pumps following an MSLB. The staff requested the licensee to evaluate this scenario. The licensee preformed the analysis by tripping the reactor coolant pumps in the affected loop at 160 seconds when the core power from re-criticality was calculated to be greatest. The reactor coolant pumps in the unaffected loop were, as before, tripped at 100 seconds because of the stability problem with RETRAN-02.

The ratio between the maximum heat flux in the core and the critical heat flux is called the departure from nucleate boiling ratio (DNBR). The DNBR is evaluated with the VIPRE-01 code. Using VIPRE-01, the licensee calculated a DNBR of 3.28 when the coolant pumps in the unaffected loop are allowed to remain running. A DNBR of 3.15 is calculated when the coolant pumps in the unaffected loop were tripped at the time of maximum return to power in the core. Both these values are in excess of the minimum DNBR limit at Oconee for low reactor system pressure, which is 1.45.

The smallest value of DNBR for a large MSLB was calculated to occur when the concurrent loss of offsite power was assumed. Loss of offsite power would cause the reactor coolant pumps, as well as the main feedwater and condensate pumps, to trip. A minimum DNBR of 1.51 was calculated to occur 1.9 seconds into the event as a result of the loss of forced reactor coolant flow and the decrease in reactor system pressure. The DNBR limit at Oconee is 1.193 for the range of pressures that is calculated for this event.

The licensee evaluated the consequences of main feedwater to isolate for a spectrum of small steamline breaks. The requirement of no fuel failures was included as an acceptance criterion. Using RETRAN-02, the licensee identified a break size so that automatic reactor trip would not occur on either low reactor system pressure or high reactor power. As the steam generators depressurize, the unisolated feedwater flow rate would increase. In addition, as steam is lost from the break, there would be less steam flow to the turbine, less steam flow to the feedwater heaters, and a reduced feedwater temperature. The resulting decrease in reactor system cold leg temperature would act to attenuate the neutron flux leakage leaving the reactor. Neutron flux attenuation would cause an error in the indicated value of reactor core power so that core power might exceed the normal reactor trip setting. The licensee evaluated the core DNBR and the internal fuel temperature for this event and determined that the applicable safety limits will be maintained to prevent fuel failure. In accordance with Reference 5, the licensee will modify Chapter 16 of DPC-NE-3005-P, Revision 1 to indicate that the acceptance criteria for this analysis are that no fuel damage will occur and that the offsite doses will remain within 10 percent of the 10 CFR Part 100 limits.

3.0 CONCLUSION

On the basis of its review of Revision 1 to DPC-NE-3005-P, including supplemental information provided by the licensee, the staff concludes that the licensee has adequately addressed the conditions contained in the staff's original safety evaluation (Reference 2). The methodology in DPC-NE-3005-P, Revision 1, is, therefore, approved and found acceptable for performing UFSAR Chapter 15 transient and accident analysis at Oconee. The plant analyses contained in the topical report are typical of those that will be incorporated into the UFSAR. For subsequent core reloads or other plant modifications, the licensee should justify that the analyses in the topical report bound the results that would be obtained for the new plant condition or should perform new analyses that are conservative for that purpose.

Principal Contributor: Walton L. Jensen, Jr.

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REFERENCES

1. UFSAR Chapter 15 Transient Analysis Methodology," DPC-NE-3005-P, Duke Power Company, July 1997.

2. Letter from David E. LaBarge, NRC to W. R. McCollum, Duke Energy Corporation, "Review of Updated Final Safety Analysis Report, Chapter 15, Transient Analysis Methodology Submittal-Oconee Nuclear Station, Units 1, 2, and 3," October 1, 1998.

3. Letter from M.S. Tuckman, Duke Energy Corporation, to NRC Document Control Desk, February 1, 1999.

4. Process, Enhanced, and Multiphase Heat Transfer," a Festscrift for A.E. Bergles, Begell House, 1996.

5. Letter from M.S. Tuckman, Duke Energy Corporation, to NRC Document Control Desk, May 5, 1999.