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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT DPC-NE-3005-P

UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) CHAPTER 15

TRANSIENT ANALYSIS METHODOLOGY

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated July 30, 1997 (Reference 1), supplemented by letter dated July 23, 1998 (Reference 2), Duke Energy Corporation (Duke) submitted Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," describing the methodology used by Duke for analyzing the nonloss-of-coolant accident (non-LOCA) UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station (ONS/Oconee), Units 1, 2, and 3. The objective of this topical report is to implement a revised non-LOCA transient and accident analysis methodology and establish a new licensing basis for ONS.

Duke received NRC approval to perform core reload design analyses for ONS in 1981. The set of licensing basis transients and accidents in the current UFSAR has essentially remained the same as in the original FSAR. However, future reload core designs will require reanalysis of the UFSAR Chapter 15 transients and accidents due to advanced fuel assembly designs, longer fuel cycles, increased steam generator tube plugging, and more efficient core designs. Therefore, Duke proposes to use the methodology described in this topical report to reanalyze the ONS Chapter 15 events in order to establish an up-to-date design basis, and to support advanced fuel assembly and core reload designs.

2.0 EVALUATION

The acceptability of Duke's nuclear and thermal-hydraulic analysis models and methods for simulating the ONS UFSAR Chapter 15 non-LOCA analyses is evaluated herein. Since many of these models and methods have previously been reviewed and approved by the NRC staff, the evaluation focused on any new models and methods, and on the specific application of the methods to the reanalysis of ONS transients and accidents.

2.1 RETRAN-02 One-Dimensional Kinetics Model

The RETRAN-02/MOD5.1 code was reviewed generically by the NRC and approved for use provided plant-specific methods have also been reviewed by the NRC (Reference 3). In this report (DPC-NE-3005-P), a new modeling application was made. A one-dimensional (1-D) kinetics model was used to model the core response for transients for which point kinetics does not provide sufficient results. The 1-D kinetics equations are derived from the neutron diffusion equation by assuming that the change in the radial neutron flux with time is relatively small. This model used the three-dimensional (3-D) nodal physics code, SIMULATE-3K (Reference 4), to generate the 1-D nuclear parameters (nuclear cross-sections and kinetics parameters) and a linking utility program (XGEN) to functionalize the nuclear parameters against the RETRAN-02 thermal-hydraulic feedback variables (moderator density and fuel temperature). The 1-D kinetics licensing basis analyses used the nuclear parameters (cross-sections and kinetics parameters) that would yield the same physics parameters (moderator temperature coefficient, Doppler coefficient, control rod worth, etc.), that were used in the licensing basis analyses using the point kinetics model. SIMULATE-3K was used in this limited application because of its ability to iteratively modify the nuclear parameters until the desired physics parameters are achieved. The resultant nuclear parameter modification factors were then used to generate the nuclear parameters for the RETRAN 1-D kinetics model.

Duke has presented comparisons of the RETRAN 1-D kinetics model with SIMULATE three-dimensional calculations. In general, the results indicate that with the 1-D nuclear data generated by this methodology, for any perturbation, the RETRAN 1-D kinetics model (1-D representation of the core) would predict a similar core response as the global response predicted by the SIMULATE 3-D representation of the core. The staff, therefore, finds this limited use of SIMULATE-3K acceptable. The SIMULATE-3K code was also used to calculate the effects of a control rod ejection accident, as discussed in Sections 2.5 and 2.6 below.

2.2 RETRAN-3D in RETRAN-02 Mode

The RETRAN-3D/MOD001F code is a recent version of the RETRAN code which has not been reviewed and approved by the NRC. Duke has submitted the code to the NRC for generic review in a separate licensing action. RETRAN-3D incorporates new models and equations, including additional balance equations to predict non-equilibrium phenomena and 3-D core kinetics, as well as advanced numerical solution schemes and correlations. However, in this report (DPE-NE-3005-P), Duke has not included any of the non-equilibrium or 3-D core modeling techniques. The application of RETRAN-3D was limited to the "RETRAN-02 mode" which is intended to replicate RETRAN-02. Only the advanced solution scheme and correlations of RETRAN-3D were utilized. This limited application of RETRAN-3D was used to analyze the startup accident, loss of flow, locked rotor, and turbine trip events and the results were compared to those obtained with the NRC-approved RETRAN-02 code. Based on the good agreement between the results of RETRAN-02 and RETRAN-3D in the RETRAN-02 mode for these transients discussed below, the staff concludes that this limited use of the RETRAN-3D code is acceptable.

2.3 VIPRE-01 Additional Features

The VIPRE-01/MOD2 code is used for steady-state and transient core thermal-hydraulic analyses and has been approved by the NRC for ONS licensing calculations (Reference 5). The version used in the DPC-NE-3005-P safety analyses is designated as VIPRE-01/MOD2F, and incorporates several additional features. However, the constitutive equations, correlations, and solution schemes of the VIPRE-01/MOD2 code have been preserved. The following changes were incorporated into VIPRE-01/MOD2F:

(1) The BWC, BWCMV, and BWU-Z critical heat flux (CHF) correlations were added.

The NRC Safety Evaluation Report for VIPRE-01 states that whenever Duke intends to use other CHF correlations, power distributions, fuel pin conduction models or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, Duke must submit justification of these changes for NRC review and approval. The B&W BWC correlation with VIPRE-01 has been approved by the NRC with the design limit of 1.18 (Reference 6). Use of the BWCMV correlation has been approved by the NRC for use with VIPRE-01 with a design limit of 1.21 (Reference 7). Use of the BWU-Z correlation with a design minimum DNBR limit of 1.19 with the Mk B11V fuel design is currently under review by the NRC staff in a separate licensing action and preliminary indications are that it will be approved for use with VIPRE-01 (Reference 8). Therefore, the incorporation of the BWC, BWCMV, and BWU-Z CHF correlations into VIPRE-01/MOD2F is acceptable, provided these correlations are used over their approved ranges of applicability.

(2) An option to allow use of either a linear interpolation or a spline fit for the input nodal axial power profile was added.

The spline fit option was originally incorporated to replace inadequacies in the linear interpolation routine. Because of straight line interpolation from point to point, linear interpolation did not conserve area under the curve and therefore would tend to under predict the axial shape uniformly, which is nonconservative for departure from nucleate boiling (DNB) calculations. However, in order to be able to duplicate previous analyses that used linear interpolation, both options were incorporated. The staff finds this acceptable.

(3) An option to allow input of the power hot channel factor and the local heat flux hot channel factor to a subchannel for calculating the DNB ratio (DNBR) in that subchannel was added.

The local heat flux hot channel factor has not been applied in Oconee DNBR analyses beginning with the Oconee Unit 1 Cycle 14 reload. The power hot channel factor has been applied during the entire operating history of Oconee and is not new to the Oconee licensing basis. Since there is no difference being introduced in the proposed methodology, the option is acceptable.

(4) An enhanced iteration scheme was added.

The logic in the original VIPRE-01/MOD2 code did not always cause the iteration to converge when the input parameter yielded a minimum DNBR value significantly different from the target minimum DNBR limit and would at times also cause the iteration process to stall. Logic was therefore added to improve the iteration technique to circumvent these problems. The staff finds the enhanced iteration logic acceptable.

2.4 ARROTTA Code for Rod Ejection Analysis

The core neutronic response to a control rod ejection event was calculated with the Electric Power Research Institute ARROTTA code (Reference 9). ARROTTA is a 3-D, 2-energy group diffusion theory code which uses neutron cross sections, discontinuity factors, and 6 groups of delayed neutron precursor data generated with CASMO-3 (Reference 10). The ARROTTA time-dependent core power distribution is used as input to the subchannel core thermal-hydraulics analysis performed with VIPRE-01. VIPRE-01 calculates the fuel temperatures, the allowable power peaking to avoid exceeding the DNBR limit, and the core coolant expansion rate. The allowable power peaking is then used along with a post-ejected condition fuel pin census to determine the percentage of pins exceeding the DNB limit. The coolant expansion rate is input to a RETRAN-02 model of the reactor coolant system to determine the peak pressure resulting from the core power excursion. Duke has indicated that ARROTTA is only used for the rod ejection accident and, because of the rapid nature of this event, the neutronics solution rather than the moderator feedback effects are most important for this application. The NRC has approved the use of ARROTTA by Duke for rod ejection analysis for the McGuire and Catawba Nuclear Stations (Reference 11) and the use of ARROTTA in DPC-NE-3005-P is consistent with the previously approved methodology. Therefore, the staff concludes that ARROTTA is acceptable for the Duke analysis of the rod ejection event in Oconee.

2.5 SIMULATE-3K Code for Rod Ejection Analysis

The SIMULATE-3K code (Reference 4) was also used by Duke to calculate the core power response and three-dimensional power distribution resulting from a control rod ejection event. SIMULATE-3K is a three-dimensional transient neutronic version of the NRC-approved SIMULATE-3P code and utilizes the same neutron cross section library. It employs a fully implicit time integration of the neutron flux, delayed neutron precursors, and heat conduction models. Code validation was performed by the code vendor (Studsvik of America, Inc.) during development of SIMULATE-3K. The validation included benchmarks of the fuel conduction and thermal-hydraulic model, the transient neutronics model, and the coupled performance of the transient neutronics and thermal-hydraulic models. Duke comparisons of SIMULATE-3K with ARROTTA for the Oconee rod ejection analysis presented in DPC-NE-3005-P indicate very good agreement for core power versus time for the ejection occurring at end-of-cycle from the maximum allowable power level with three and four reactor coolant pumps (RCPs) operating and from both beginning-of-cycle and end-of-cycle hot zero power conditions. Although larger deviations occur for the ejection event initiated at beginning-of-cycle at the maximum allowable power level with three and four RCPs operating, the comparisons indicate that SIMULATE-3K results in more conservative values than ARROTTA. Therefore, the staff concludes that SIMULATE-3K is acceptable for the Duke analysis of the rod ejection event in Oconee.

2.6 Chapter 15 Safety Analyses

The core thermal-hydraulic analysis for most of the transients considered in this topical report is based on the NRC-approved statistical core design (SCD) methodology (Reference 12). This methodology includes uncertainties on the initial conditions of core average power, core inlet temperature, core exit pressure, and core inlet flow. Therefore, when performing an SCD analysis, initial condition uncertainties are not included in these four parameters as they are already included in the SCD DNBR limit. When non-DNB analyses are being performed, the uncertainties are included. For either type of analysis, the uncertainty in the timing of a particular action is accounted for by uncertainty-adjusting the actuation setpoints.

Reactivity and Power Distribution Anomalies

Six transients are considered in this category: (1) startup accident, (2) control rod withdrawal at power, (3) moderator dilution event, (4) cold water event, (5) control rod misalignment, and (6) control rod ejection accident.

The startup accident involves a reactivity addition due to an uncontrolled control rod withdrawal from a subcritical or low power condition. The current licensing basis for this event is that the reactor thermal power does not exceed 112 percent of rated thermal power (RTP) and reactor coolant system (RCS) pressure does not exceed code pressure limits. A high flux level and a high pressure trip are assumed. The proposed new acceptance criteria for this event are that the peak RCS pressure remains below 110 percent of the design pressure of 2500 pounds per square inch gauge (psig) and that no fuel failures result as demonstrated by not exceeding the DNBR limit. This is consistent with Section 15.4.1 of the NRC Standard Review Plan (SRP). The RETRAN-02 code is used to determine the peak transient primary system pressure. If the peak heat flux for the peak primary pressure analysis exceeds the maximum allowed steady-state value, the VIPRE-01 code is used to calculate the minimum DNBR for the transient using the SCD methodology. The core power distribution is analyzed with the SIMULATE-3P code. The event models three RCPs in operation and a maximum reactivity addition rate. Reactor trip is expected to occur on high pressure, high power, or the flux/flow imbalance trip functions.

The rod withdrawal event initiates from an accidental withdrawal of a control rod group while the reactor is at power. The current licensing basis is that the reactor thermal power does not exceed 112 percent of RTP and RCS pressure does not exceed code pressure limits. A high reactor coolant outlet temperature trip, a high reactor coolant system pressure trip, and a high power level (neutron flux level) trip are available to terminate this event. The proposed new acceptance criteria for this event require that the peak RCS pressure remains below 110 percent of design pressure and that the DNBR limit not be exceeded. This is consistent with Section 15.4.2 of the NRC SRP. The peak primary pressure case is analyzed with the RETRAN-02 code and VIPRE-01 is used to calculate the minimum DNBR. The core power distribution is analyzed with the SIMULATE-3P code. It is expected that the thermal-hydraulic conditions at the limiting DNB statepoint will be within the ranges covered by the SCD methodology. Reactor trip is expected to occur on high flux or high pressure, although the high coolant temperature and flux/flow/imbalance trips may also provide protection.

A RCS moderator dilution event occurs when the soluble boric acid concentration of makeup water supplied to the RCS is less than the concentration in the existing reactor coolant. Because Oconee's operating license was issued prior to the NRC SRP, the current licensing basis analyzes the event only from rated power (Mode 1) and refueling (Mode 6) conditions and requires that reactor thermal power not exceed 112 percent of RTP, RCS pressure not exceed code allowable limits, and a minimum shutdown margin of 1 percent $\Delta k/k$ be maintained. The proposed new acceptance criteria are based on manual operator action at least 15 minutes during power operation and at least 30 minutes during refueling following actuation of the alarm credited for alerting the operator of a moderator dilution occurring. This ensures that the event is terminated by the operator before the DNBR limit or the peak primary pressure limit is violated and is consistent with Section 15.4.6 of the NRC SRP. For reload evaluations, the cycle-specific highest critical boron concentration and the initial boron concentration closest to the critical concentration are used.

The cold water accident is initiated with an inadvertent startup of an idle RCP, which causes a reduction in moderator temperature and a power excursion due to moderator feedback effects. The current licensing basis assumes the event is initiated from 50 percent power with two RCPs operating in one loop and two idle loops. This is no longer permitted by Oconee TS, which do not allow operation while critical with less than three RCPs in operation. Therefore, these analyzed conditions currently represent a bounding case relative to allowed operating conditions. The acceptance criteria would remain, i.e., minimum DNBR does not violate the acceptance criterion and system pressure limits (110 percent of design pressure) are not exceeded. This is consistent with Section 15.4.4 of the NRC SRP. The minimum DNBR would be determined using the SCD methodology. The proposed reanalysis of this event would assume that it is initiated with an inadvertent startup of a fourth RCP from an initial three-pump operating condition using the two-loop RETRAN-02 model.

The most limiting control rod misalignment event is the dropped rod since it presents the greatest potential for violating the minimum DNBR or system pressure limits. Although the event initially causes a rapid reduction in power and moderator temperature, the negative moderator temperature coefficient would subsequently result in a power increase. If the reactor is operating with the Integrated Control System (ICS) in automatic, rod withdrawal by the ICS will add to the increase in power. Although the analog ICS is being replaced by a digital ICS, the same modeling philosophy is retained in the analysis of this event. The transient response is analyzed with the RETRAN-02 code and the DNB analysis is performed with VIPRE-01 using the SCD methodology. The core power distribution is analyzed with the SIMULATE-3P code. The acceptance criteria remain that RCS pressure does not exceed 110 percent of design pressure and minimum DNBR does not violate the DNBR limit. This is consistent with Section 15.4.3 of the NRC SRP.

In addition to the dropped rod event, Duke has evaluated the misalignment event where a control rod is misaligned from the remainder of the rods in its bank. Since this may produce an increase in core peaking which decreases the margin to DNB, SIMULATE-3P was used to confirm that the asymmetric power distribution resulting from the rod misalignment will not result in DNB.

The rod ejection event is initiated by a failure of a control rod drive mechanism housing, which allows a control rod to be rapidly ejected from the core by the RCS pressure differential. The licensing basis criteria are that the accident will not further damage the RCS and that the offsite dose will be within the 10 CFR Part 100 limits. The first criterion is met by demonstrating that the peak fuel enthalpy remains below 280 cal/g. The proposed reanalysis would also require the peak primary pressure to remain within the Service Limit C as defined in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (120 percent of the 2500 psig design pressure, or 3000 psig). The core power excursion is simulated with the ARROTTA or SIMULATE-3K 3-D transient code. VIPRE-01 is used to calculate the fuel temperatures, the allowable power peaking, and the coolant expansion rate. The allowable power peaking is used along with a post-ejection fuel pin census to determine the percentage of fuel pins exceeding the DNBR limit. All pins in DNB are assumed to experience clad failure for dose calculational purposes. The coolant expansion rate is input to RETRAN-02 to determine the peak pressure. As mentioned in Sections 2.4 and 2.5 above, ARROTTA and SIMULATE-3K are acceptable codes for analyzing the rod ejection accident in Oconee.

The staff concludes that the appropriate reactivity and power distribution anomalies will be reanalyzed using acceptable methods and that current licensing basis acceptance criteria will remain in effect or will be updated to conform to NRC SRP acceptance criteria.

Loss of Coolant Flow

The loss of coolant flow transient could be initiated by an electrical failure to the RCPs and result in one or more RCP coast downs. Either the pump monitor trip or the flux/flow/imbalance trip are used for reactor protection during this event. This event is classified as an event of moderate frequency. The acceptance criterion for this event is that no fuel failures will result as demonstrated by not exceeding the DNBR limit established at Oconee.

There are five bounding scenarios selected for the analysis of this transient including various numbers of RCP(s) coast downs from initial four or three RCP operation. RETRAN-02 and VIPRE-01 are used for thermal-hydraulic analyses and DNBR calculations. Initial and boundary conditions assumed in the analysis are conservatively selected for limiting consequences of the event. The results of the analysis confirmed that the transient minimum DNBR is approximately 1.68, which is sufficiently above the allowable minimum DNBR of 1.3 at Oconee. The staff finds this analysis methodology acceptable.

Locked Rotor

The locked rotor accident is the result of an instantaneous seizure of one RCP due to mechanical failure of the RCP. The analysis of this event assumes that the off-site power is available during this accident which is consistent with the current Oconee UFSAR. The flux/flow/imbalance trip is used for reactor protection during this event. The proposed licensee acceptance criteria include (1) fuel failure should be sufficiently limited to maintain core cooling capability, and (2) radiological consequences should be within the 10 CFR Part 100 guidelines. Since this event will also cause heatup of the RCS, in response to the staff request, the licensee has committed to include a peak RCS pressure acceptance criterion of 110 percent of design pressure for the locked rotor accident (Reference 2). The results will be included in the UFSAR.

There are three scenarios analyzed for this event including one locked rotor from four pump operation and one locked rotor in the loop either with the idle RCP or without idle RCP from three-pump operation. RETRAN-02 and VIPRE-01 are used for thermal-hydraulic analysis and DNBR and maximum allowable radial peaking limits calculations. Initial and boundary conditions assumed in the analysis are conservatively selected for limiting consequences of the event. The results of this analysis indicate that there are no fuel failures and the acceptance criteria are met for this event. The staff finds the proposed analysis methodology acceptable.

<u>Turbine Trip</u>

The turbine trip transient can be initiated by a generator trip, low condenser vacuum, loss of lubrication oil, turbine trust bearing failure, turbine overspeed, or a manual trip. The turbine trip event could cause an increase in RCS temperature and pressure. The high RCS pressure trip is used for reactor protection during this event. This event is classified as an incident of moderate frequency. The acceptance criterion for this event is that the peak RCS pressure shall not exceed 110 percent of the design pressure. There is no DNB concern with this heatup transient.

The bounding scenario analyzed for the event is that of a turbine trip at full power under four RCP operating conditions. RETRAN-02 and VIPRE-01 are used to calculate the thermal-hydraulic responses. Initial and boundary conditions assumed in the analysis are conservatively selected for limiting consequences of the event. The results of this analysis confirm that the peak RCS pressure during the transient is within the acceptance criterion with sufficient margin. The staff finds that the method of analysis is acceptable.

<u>Steam Generator Tube Rupture</u>

The Steam Generator Tube Rupture (SGTR) analysis documented in the current Oconee UFSAR assumes no operator action at the beginning of the event and a low RCS pressure trip in about 8 minutes. The methodology proposed in the topical report assumes operator action to identify that a tube rupture has occurred and to manually trip the reactor in 20 minutes. Also, immediate action to maximize emergency core cooling system injection is needed.

The staff will require the licensee to modify the proposed methodology for the SGTR analysis to be consistent with the licensing bases established in the UFSAR. We will provide our safety evaluation of this item after the issue is resolved.

<u>Large Steamline Break</u>

The licensee is modifying the analysis methodology for this event. The new methodology will assume no main feedwater isolation during the event, which is consistent with the current Oconee UFSAR. The staff will provide its safety evaluation of this item after the issue is resolved.

The split reactor vessel modeling approach was approved by the NRC for the Oconee containment mass and energy release analysis methodology (Reference 13) and is similar to the method approved by the NRC for the McGuire and Catawba steamline break analysis (Reference 11). Although some differences exist to conservatively model the core response as compared to the mass and energy release, the staff finds the modeling acceptable for Oconee steamline break analyses.

Small Steamline Break

The licensee is modifying the analysis methodology for this event. The new methodology will assume no main feedwater isolation during the event. Also, an acceptance criterion will be added to require no fuel failure (no DNB) during this event since it is an incident of moderate frequency. The staff will provide its safety evaluation of this item after the issue is resolved.

3.0 CONCLUSIONS

Duke Topical Report DPC-NE-3005-P and its supporting documents, including the Duke responses to the NRC request for additional information, were reviewed to determine the acceptability of the revised non-LOCA transient and accident analysis methodology that will establish a new licensing basis to be used for future Oconee Chapter 15 analyses. Since most of the models and methods have been previously reviewed and approved by the NRC, the review focused on any new models and methods as well as on the specific application of the methods to the reanalysis of transients and accidents. Except for the following, the Duke methodology, as documented in DPC-NE-3005-P was found to be acceptable.

The licensee has committed to inclusion of a peak RCS pressure acceptance criterion of 110 percent of design pressure for the locked rotor event (Reference 2). The staff requires that the proposed SGTR methodology be modified to allow no operator action at the beginning of the event and initiation of a low RCS pressure trip in approximately 8 minutes to be consistent with the licensing bases in the UFSAR. The licensee is also modifying the proposed methodology for the large and small steamline break events to assume no main feedwater isolation during the events for consistency with the current Oconee UFSAR. In addition, since it is an incident of moderate frequency, an acceptance criterion will be added to the small steamline break analysis to require no fuel failures. The staff will provide safety evaluations of these items when the issues are resolved.

Principal Contributor: Chu-Yu Liang

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