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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT DPC-NE-3000

"THERMAL-HYDRAULIC TRANSIENT ANALYSIS METHODOLOGY"

OCONEE NUCLEAR STATIONS, UNITS 1, 2, AND 3

MCGUIRE NUCLEAR STATIONS, UNITS 1 AND 2

CATAWBA NUCLEAR STATIONS, UNITS 1 AND 2

DOCKET NOS. 50-269, 50-270, 50-287, 50-369, 50-370, 50-413 AND 50-414

1.0 INTRODUCTION

Duke Power Company (DPC) submitted Topical Report DPC-NE-3000, "The Thermal-Hydraulic Transient Analysis Methodology, Oconee Nuclear Station, McGuire Nuclear Station, and Catawba Nuclear Station" in a letter dated September 29, 1987 (Ref. 1), as revised by a letter dated May 11, 1989 (Ref. 2). Additional information was also provided in References 3 to 10. This topical report documents the development of the thermal-hydraulic simulation models for the Oconee, McGuire, and Catawba plants using RETRAN-02 and VIPRE-01 computer codes and provide DPC's responses to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions" (Ref. 11).

RETRAN-02 is a large and sophisticated computer code developed to simulate a wide spectrum of thermal-hydraulic transients for both pressurized water reactors and boiling water reactors (Ref. 12). VIPRE-01 is an open channel code designed to evaluate DNBR and coolant state for steady state and transient core thermal-hydraulic analyses (Ref. 13). Both RETRAN-02 and VIPRE-01 have been approved for PWR licensing calculations with generic limitations (Refs. 14 to 15).

Generic Letter 83-11 requests that each licensee or vendor who intends to use large, complex computer codes to perform their own safety analyses to demonstrate their proficiency to use the codes by submitting code verification performed by themselves. To demonstrate their technical competence in using the RETRAN computer code and qualify their RETRAN models for thermal-hydraulic transient simulation, DPC provided in the RETRAN portion of this topical report: (1) detailed descriptions of the plant nodalizations, control system models, code models, and code options selected for use in the analysis, (2) analyses benchmarked against start-up test data and plant operational transient data from the Oconee, McGuire, and Catawba plants.

In the SER for VIPRE-01, the staff requests each user to document and submit a separate report which, (1) describes how they intend to use VIPRE, and (2) provides justification for specific modeling assumptions, choices of particular

models and correlations, and input values of plant specific data such as turbulent mixing coefficient and grid loss coefficient. DPC previously submitted VIPRE-01 models for use in steady-state which have been addressed in specific SERs. VIPRE-01 models for transient applications are addressed in this SER.

2.0 STAFF EVALUATION

The staff review and evaluation of the Topical Report DPC-NE-3000 addresses: (1) DPC's competence in using the RETRAN and VIPRE computer codes, (2) the degree to which the topical report and supplemental information satisfy requirements in the VIPRE-01 and RETRAN SERs; and, (3) the ability of the RETRAN simulations to match plant operational data. The review of this topical was performed with technical assistance from International Technical Services, Incorporated (ITS) and its review findings are contained in the Technical Evaluation Report (TER) which is attached. The staff has reviewed the TER and concurred with its findings.

3.0 FINDINGS AND CONCLUSIONS

The staff has reviewed the Topical Report DPC-NE-3000, which documents the development of the thermal-hydraulic simulation models for the Oconee, McGuire, and Catawba plants. Overall we conclude that the licensee has demonstrated a high degree of technical competence in using RETRAN-02 and VIPRE-01 computer codes. Specific findings and conclusions regarding the RETRAN and VIPRE models are discussed below.

RETRAN FINDINGS

We find that DPC's RETRAN-02 models to be acceptable for the simulation of the symmetric non-LOCA thermal-hydraulic transients for the McGuire, and Catawba Nuclear Units, subject to the limitations listed below. However, the RETRAN-02 models for Oconee have not been shown to be adequate for best estimate nor licensing calculations, and are therefore not approved for either of these applications.

(1) With respect to analyzing transients which result in a reduction in steam generator secondary water inventory, use of the RETRAN-02 steam generator modeling is acceptable, only for transients in that category for which the secondary side inventory for the effective steam generator(s) relied upon for heat removal never decreases below an amount which would cover enough tube height to remove decay heat.

(2) All generic limitations specified in the RETRAN-02 SER (Reference 14).

VIPRE FINDINGS

We find that the subject topical report, together with DPC responses, contains sufficient information to satisfy the VIPRE-01 SER requirement that each VIPRE-01 user submit a document describing proposed use, sources of input variables, and selection and justification of correlations as it relates to use by DPC for FSAR Chapter 15 analyses regarding Oconee, McGuire and

Catawba. We further find that the manner in which the code is to be used for such analyses, selection of nodalization, models, and correlations provides, except as listed below, adequate assurances of conservative results and is therefore acceptable. Furthermore, the use of the DPC developed statistical core design methodology as approved in the Staff Safety Evaluation Report on DPC-NE-2004, is approved for the transient application subject to the same conditions.

The following items are limitations regarding VIPRE-01 application presented in DPC-NE-3000 and its supplemental materials:

(1) Determination of acceptability is based upon review of selection of models/correlations for transients involving symmetric core neutronic and thermal-hydraulic conditions only. Thus, the VIPRE-01 models are approved for use in analyzing symmetric transients only;

(2) When using the DPC developed SCD method, the licensee must satisfy the conditions set forth in the staff's safety evaluation of DPC-NE-2004;

(3) Whenever DPC intends to use other CHF correlations, power distribution, fuel pin conduction model or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, DPC must submit its justification for NRC review and approval;

(4) Core bypass flow should be determined on cycle-by-cycle bases;

(5) All generic limitations specified in the VIPRE-01 SER.

4.0 REFERENCES

1. Letter from H. B. Tucker (DPC) to USNRC, "Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Response to Generic Letter 83-11," September 29, 1987.

2. Letter from H. B. Tucker (DPC) to USNRC, Attachment "DPC-NE-3000 Revision 1," May 11, 1989.

3. Letter from H. B. Tucker (DPC) to USNRC, Attachment "Duke Power Responses to NRC Questions Dated April 7, 1989 Regarding DPC-NE-3000," June 15, 1989.

4. Letter from H. B. Tucker (DPC) to USNRC, "Response to Questions Regarding Differences Between Duke Topical Reports DPC-NE-2003 and DPC-NE-3000," June 19, 1989.

5. Letter from H. B. Tucker (DPC) to USNRC, "Duke Power Response to NRC Questions Regarding Steam Generator Heat Transfer Modeling with the RETRAN Code," August 9, 1989.

6. Letter from H. B. Tucker (DPC) to USNRC, Attachment 1 "Responses to NRC Questions on the McGuire/Catawba Sections of DPC-NE-3000" and Attachment 2 "Revisions to Section 4 of DPC-NE-3000," September 13, 1989.

7. Letter from H. B. Tucker (DCP) to USNRC, "Response to NRC Questions on DPC-NE-3000 Dated July 25, 1989," February 20, 1990.

8. Letter from M. S. Tuckman (DPC) to USNRC, "Supplemental Information to Assist in Review of Topical Reports DPC-NE-3000 and DPC-NE-2004," August 29, 1991.

9. Letter from H. B. Tucker (DPC) to USNRC, "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

10. Letter from H. B. Tucker (DPC) to USNRC, "Final Response to Questions Regarding the Topical Reports Associated with the M1C8 Reload Package," November 5, 1991.

11. Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (Generic Letter No. 83-11), USNRC, February 8, 1983.

12. "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM Revision 2, EPRI, November 1984.

13. "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI NP-2511-CCM Revision 2, EPRI, July 1985.

14. Letter, C. O. Thomas (NRC) to T. W. Schnatz (UGRA), September 4, 1984, (Transmittal of RETRAN-02 Safety Evaluation Report).

15. Letter, C. E. Rossi (NRC) to J. A. Blaisdell (UGRA), May 1, 1986, (Transmittal of VIPRE-01 Safety Evaluation Report).

16. "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003, August 1988.

Date: November 15, 1991

TECHNICAL EVALUATION OF THE THERMAL-HYDRAULIC TRANSIENT ANALYSIS METHODOLOGY TOPICAL REPORT DPC-NE-3000 FOR THE DUKE POWER COMPANY OCONEE, MCGUIRE AND CATAWBA NUCLEAR STATIONS Part 1

1.0 INTRODUCTION

DPC-NE-3000, dated July 1987 (Ref. 1), documents results of a series of studies performed by Duke Power Company (DPC) to support the development of thermal-hydraulic transient analysis methods and provides DPC's response to Generic Letter 83-11 (Ref. 2). These methods were developed using the RETRAN-02 (Ref. 3) and VIPRE-01 (Ref. 4) computer codes, both of which have been approved, subject to conditions (Refs. 5 & 6). The stated objective of the subject report was for DPC to demonstrate DPC capability and technical competence through RETRAN analysis of its Oconee plants (which are B&W plants) and its McGuire and Catawba plants (which are Westinghouse plants).

The purpose of this review, which is based upon a review of the submitted materials (Refs. 1, 7-13), is to determine (i) the degree of DPC's technical competence demonstrated in the transient analyses, (ii) acceptability of the RETRAN plant models by review of the accuracy of the results obtained using the computer codes and submitted models and (iii) adequacy of DPC's documentation of its VIPRE-01 models to fulfill VIPRE SER requirements.

This technical evaluation report (TER) is divided into two parts: Part 1 presents our evaluation (in accordance with the RETRAN SER) of DPC's use of the RETRAN computer code and the acceptability of the DPC RETRAN models for Oconee and McGuire/Catawba plants; Part 2 contains an evaluation (in accordance with the VIPRE SER) of DPC's intended method for use of the VIPRE computer code in transient application for the same plants.

2.0 EVALUATION

Acceptability of DPC's application of the RETRAN computer code for thermal-hydraulic calculations of the transient behavior of its Oconee and McGuire/Catawba plants is discussed below.

2.1 Oconee Plant Model

DPC developed two Oconee RETRAN models: (1) a one-loop plant model to be used where there is little asymmetry between loop responses and (2) a two-loop plant model to be used when asymmetric conditions are expected in the analysis. Detailed descriptions of the plant nodalizations and models selected for use in the analysis are presented in Chapter 2 of the topical report.

In the one-loop model, DPC models both steam generators and the accompanying hot and cold legs by one hot leg, one once-through steam generator (OTSG) and one cold leg. The core and steam generator nodalizations are the same as those in the two-loop plant model.

The base two-loop Oconee plant model consists of two separate loops each containing one hot leg, an OTSG and two cold legs. The OTSG is nodalized with equal height shell and tube side volumes except at the bottom of the steam generator. DPC stated that the specific degree of detail selected (i.e. the number of nodes) for the OTSG is necessary to model the void distribution in the OTSG.

The modeling of OTSGs is very difficult because in normal operation the steam in the upper portion of the SG is super heated and the SG tubes are partially uncovered (in marked contrast to U-tube type plants). Therefore the primary-to-secondary heat transfer rate is a function of the two-phase mixture height on the secondary side and the predicted transient behavior is strongly dependent upon the two-phase modeling on the secondary side of the steam generator. The mixture interface location and its transient behavior are

very difficult to model with RETRAN, facts which DPC has acknowledged (Ref. 10).

DPC used the non-equilibrium pressurizer option to model the pressurizer (PZR) for best-estimate safety analysis. Other model options are used as necessary to obtain conservative results for Chapter 15 type analyses.

Although the DPC responses to NRC questions referred to their experience with three sets of nodalizations and to sensitivity studies performed to arrive at the base nodalization, DPC did not justify selection of built-in RETRAN thermal-hydraulic models and correlations. Furthermore, although the nodalization study indicates that the model was converged, it did not indicate accurate convergence to the mixture level on the secondary side.

In addition, DPC presented qualitative arguments supporting the selection of various nodalizations for other plant components (such as the reactor vessel) and the selection of the use of the certain models such as the bubble rise model and the non-equilibrium model.

The Oconee base model is based on the Unit 1 thermal design flow, since it is lower than Units 2 and 3 and is conservative with respect DNB. The RETRAN initial conditions for computed RCS flow as well as other key plant parameters were adjusted by DPC on a transient-by-transient basis to better match the plant data as noted later.

RETRAN control systems were developed and used extensively by DPC to specify transient boundary conditions, such as automatic plant actions and operator actions as well as control actions by modulating valves, changing fill rates or reactivity and simulation of trip actuation. The control system was also used to compute the steam generator level by emulating the plant measurement devise by taking DP across SG pressure taps. In addition, uncertainty in the degree of SG tube fouling generally resulted further in large discrepancies between the predicted and measured data as discussed in the following sections.

2.1.1 Oconee RETRAN Model Qualification

DPC chose to demonstrate the adequacy of the base plant model for both Oconee and McGuire/Catawba plants through comparison of RETRAN analyses to available plant data, providing reasonably thorough analyses of the transients analyzed. However, DPC provided only limited justification for its plant nodalization, input selection, and selection of particular correlations built into the code, and did not present any description of its RETRAN control systems models in the topical report. DPC took the position that the test of the model was in its ability to reproduce plant data, notwithstanding the fact that it is widely recognized that modeling of a once-through steam generator is difficult with RETRAN (as is evident from results of DPC benchmark analysis).

Therefore, this evaluation is based upon a review of the ability of the base model (best-estimate model) to benchmark startup test data and several operational transient data over a wide range of plant conditions.

2.1.2 Benchmark Analyses

For the purpose of benchmarking the base Oconee RETRAN models, DPC analyzed 11 tests and transients, one of which was a transient which occurred at Arkansas Nuclear One - Unit 1, a sister plant.

The one-loop model was used for six analyses: (1) Loss of Main Feedwater, (2) Turbine Bypass Valve Failure Following Reactor Trip, (3) Loss of Offsite Power, (4) Steady State Natural Circulation Comparisons, (4) Control Rod Group Drop, and (6) Main Feedwater Pump Trip.

The two-loop model was used in the five remaining analyses: (1) Steam Generator Overfeed Following Reactor Trip, (2) Overcooling Following Loss of ICS Power, (3) Reactor Coolant Pump Coastdowns, (4) Turbine Bypass Valve Failure, and (5) Reactor Trip from Three Reactor Coolant Pump Operation.

2.1.2.1 Loss of Main Feedwater

DPC analyzed the loss of main feedwater event which occurred in August 1984 at Oconee Nuclear Station Unit 3 while it was operating at full power. Letdown was manually isolated in the first 10 seconds and RCS makeup flow was increased manually. Only one high pressure injection (HPI) pump operated during the transient. All three emergency feedwater (EFW) pumps started immediately following the loss of the MFW pumps, and contributed to maintaining SG levels.

The modeling of this transient was revised and resubmitted by DPC (Ref. 8) to better model the boundary conditions. In the revised analysis, RCS flow and T-ave were adjusted to match the plant data which resulted in T_{hot} and T_{cold} being initialized at slightly different values. Since the EFW flow data were unavailable, DPC inferred the EFW flowrate from the SG levels and used a RETRAN control system to simulate throttling of EFW to match the simulated SG level with the plant data.

DPC provided a thorough analysis for this transient. Following the trip, results indicated a modest over-prediction of primary-to-secondary heat transfer and, after the PZR spray setpoint was first reached at roughly 450 seconds into the transient, the predicted RCS pressure cycled at approximately double the frequency of the plant data until about 900 seconds into the transient. Thereafter until the EFW flow was reestablished, the predicted pressure cycled at only a slightly higher frequency than the data. During the period between the beginning of the transient and about 1100 seconds, DPC computed the PZR level to be lower than the data, and the predicted hot and cold leg (average since one loop represents both cold legs) temperatures were predicted to be lower than the plant data implying an overprediction of the primary-to-secondary heat transfer.

After reestablishment of the EFW, the computed average cold leg temperature matched the loop B data but was roughly 10 degrees above the loop A data, the computed hot leg temperatures agreed with the data and the predicted

pressurizer level was about 35 inches higher than the measured data, while the RCS pressure was predicted to decrease at roughly twice the measured rate.

DPC explained these modest differences between the predicted and plant data as being due to several factors:

(1) the code tended to couple too closely between SG temperature and RCS temperature during low SG flow conditions (which were present during the EFW stages of this transient) due, in part, to overprediction of the boiling length/mixture level caused by the lack in RETRAN of an unequal phase velocity model in the SG tube region;

(2) the overprediction of pressure decrease following reestablishment of EFW at 1310 seconds was due to pressurizer modeling which did not model the expected stratification of fluid which would accompany an insurge of cooler primary loop water, which would affect pressure response during the outsurge which accompanied the renewed EFW flow; and

(3) the lack of accurate modeling in RETRAN of interphase heat transfer which was very important in modeling the impact of pressurizer spray.

The last two factors may also have been at responsible for the facts that the PZR spray was predicted to cycle twice as frequently as the data during the period between 450 to 900 seconds and at approximately the same rate between 900 to 1300 seconds.

2.1.2.2 Turbine Bypass Valve Failure Following Reactor Trip

The turbine bypass valve failure occurred after an anticipatory reactor trip occurred on a main turbine trip signal. The failure was due to a malfunction in the turbine bypass system. Letdown was manually isolated in the first 10 seconds and makeup flow was increased by manually opening a second makeup valve. Main feedwater remained available throughout the event. The turbine

bypass valves were manually closed.

DPC adjusted the RCS and MFW flows to obtain the measured primary and secondary temperatures.

The DPC analysis predicted the global trend for the key plant parameters. However, the fine structure of this transient was not well predicted, with the RCS temperatures and RCS pressure being consistently underpredicted. With RCS temperatures and pressure consistently underpredicted, the PZR level would be expected to also be consistently underpredicted. However, the PZR level was overpredicted at times and underpredicted at other times during this transient. DPC stated that the "RCS pressure data ... may not be ... accurate".

An offset in the SG level, developed between the predicted results and the data after the first 10 seconds of the transient, was attributed to fouling of the SGs, causing SG level data to be in error. RETRAN did not predict the repressurization of SG pressure after the TBVs closure. DPC stated that this may be due to discrepancy in SG secondary side inventory and primary-to-secondary heat transfer rate. In addition, DPC stated that changes in slope in the data may be due to lifting and reseating of the main steam relief valves.

2.1.2.3 Steam Generator Overfeed Following Reactor Trip

Following the turbine trip, due to an Integrated Control System failure, the MFW pumps did not run back properly, resulting in overfeeding the steam generators. This led to a pump trip on high level in SG "A".

For this analysis DPC matched the initial SG levels to the plant data since the fouling in the SG was deemed less significant at the time of this transient.

The computed values of the key predicted plant parameters agreed well with the plant data. Steam generator secondary side mixture levels on the other

hand, after starting at the same levels, drifted apart (RETRAN predicting lower than the data) to maintain about the same offset after 40 seconds into the transients. DPC attributed this discrepancy to overprediction of primary-to-secondary heat transfer, which was consistent with the moderate underprediction of primary temperatures.

2.1.2.4 Overcooling Following Loss of ICS Power

Due to a spurious low hotwell level signal, Oconee Unit 3 tripped the hotwell pumps at 99% full power operation. At 73 seconds the power supply to the ICS was lost for a period of 150 seconds. During the same period, the turbine bypass valves failed at an unknown partially open position. This resulted in a loss of SG pressure control and overcooling.

DPC specified as boundary conditions the reactor power runback, MFW flow data, EFW and HPI actuation, a post-trip auxiliary steam demand, and the steam relief flowrate through a turbine control system.

Since so much was unknown after the ICS was lost and due to the partially stuck opened turbine bypass valves at an unknown position, only the first 73 seconds of this analysis was reviewed.

The repressurization of the RCS beginning at about 30 seconds was overpredicted by the code by 150 psi and the PZR level was slightly overpredicted. This was attributed by DPC to be due to the code's neglect of heat transfer between the steam and liquid regions of the PZR during the compression of the steam which accompanies the insurge.

The cold leg temperature increases were similarly overpredicted by the code during this same period, indicating underprediction of primary-to-secondary heat transfer, which was consistent with the underprediction of SG levels, and was probably also related to minor imprecisions in the modeling of power runback during the first 55 seconds of the transient.

2.1.2.5 Loss of Offsite Power at Arkansas Nuclear One - Unit 1

While operating at 100% power, Arkansas Nuclear One Unit 1 experienced a loss of offsite power. Stable natural circulation was established and maintained for more than one hour before the offsite power was restored.

DPC used as boundary conditions a one second MFW flow coastdown, EFW and HPI flows, ANO-1 MSSV lift setpoints and SG pressure vs. time control.

Although primary and secondary pressures were well predicted, the hot-to-cold leg delta T was overpredicted by nearly a factor of 2 at around 100 seconds, which was the time that natural circulation flows were set up. This implies that this analysis did not predict natural circulation flows very well. However, by roughly 150 seconds, the prediction nearly matched the data, implying a much better computation of natural circulation at this stage. DPC attributed the mismatch in the RCS temperatures during the early portion of the transient to differences in the predicted RCS flow and the actual flow during the coastdown.

2.1.2.6 Reactor Coolant Pump Coastdowns

A series of RCP coastdown tests were conducted as part of the startup testing. All of these tests were performed at hot zero power conditions considering all possible numbers of pumps available.

For this analysis, DPC stripped the two-loop model to include only those components pertinent to the benchmark remained.

The predicted and test results compared favorably, except in cases where reverse flow thorough the pump(s) occurred. For these cases, discrepancies ranged from 10 to 20% of full flow. DPC stated that where the divergent results were obtained, the divergence was in part due to suspect plant data. In addition, other discrepancies were said to occur for operating regimes in quadrants in which relatively little test data had been obtained, and therefore to not be necessarily indicative of code errors.

DPC further stated that the pump coastdown cases are not limiting with respect to the plant operating limits and that DPC does not perform transient analyses to determine operating limits with pump coastdown flow rates which are non-conservative with respect to plant data.

2.1.2.7 Steady State Natural Circulation Flow Comparisons

The RETRAN predictions were compared to calculated natural circulation flow rates from various tests and events at lowered-loop 177 fuel assembly B & W units at the end of a loss of offsite power simulation. Predictions varied from data by as much as a factor of two, with RETRAN consistently overpredicted the RCS natural circulation flows, a result which is consistent with the observed results of the ANO-1 analyses discussed above.

DPC attributed these discrepancies to prediction of a higher mixture level in the secondary side of the steam generators due to the lack of an unequal phase velocity model in RETRAN.

2.1.2.8 Control Rod Group Drop

The Group 6 control rods dropped when Oconee Unit 1 was operating at 100% power.

RCS makeup flow, MFW flow and steam generator pressure control are among the boundary conditions specified for the analysis. DPC increased the PZR surge line loss coefficient by a factor of 5 over its nominal value for analysis of this transient.

The computed plant parameters exhibited the same trend as those measured during the event. DPC stated that the increase in surge line loss coefficient was necessary to accurately model strong outsurges.

2.1.2.9 Main Feedwater Pump Trip

The 1B MFW pump tripped on low hydraulic oil pressure at Oconee Unit 1.

DPC used the reactor and turbine control valve controls, Unit Load Demand signal to the reactor control and MFW flows as boundary conditions.

The computed RCS temperatures were underpredicted slightly due to the overprediction of primary-to-secondary heat transfer, otherwise the computed key parameters agreed reasonably with those measured during the event.

2.1.2.10 Turbine Bypass Valve Failure

Following an increase in the steam generator "A" pressure signal at 100% full power at Oconee Unit 1, the turbine bypass valves opened. The erroneous pressure signal increased by 128 psi in 8 seconds, with the turbine bypass valves opening ~80%, while the actual SG pressure decreased ~25 psi during this period. After 14 seconds the erroneous SG pressure signal decreased and the bypass valves closed.

The boundary conditions used by DPC were reactor and turbine control, SG pressure signal to the turbine bypass controller, MFW flow and a reduction in the turbine bypass valve setpoint.

Since the main steam pressure response was not well predicted, the balance of the plant parameters were not well predicted.

2.1.2.11 Reactor Trip from Three Reactor Coolant Pump Operation

Oconee Unit 3 was operating at 74% full power with the B2 RCP secured. A component failure within the ICS caused a reduction in FW flow to the "A" SG. After 23 seconds, the reactor tripped on high RCS pressure.

SG levels were initially matched, but the SG "A" pressure was much higher than the data. The boundary conditions specified by DPC for the code were control rod movement, kinetics parameters, RCS makeup flow, MFW flow and SG pressure control.

The predicted RCS pressure dropped more than 100 psi below the drop in the data, and was attributed to overprediction of primary-to-secondary heat transfer due to inaccurate steam generator modeling.

2.1.3 Summary

In its modeling of the Oconee plant transient results, difficulties in accurately modeling primary-to-secondary heat transfer with the RETRAN two-phase flow and heat transfer models were a consistent source of erroneous computations (discrepancies between the predicted and measured data). These secondary-side originated difficulties also caused errors in primary-side results.

In addition, the RETRAN results consistently indicated inaccurate modeling of natural circulation flow.

Furthermore, DPC has observed that the pressurizer surge line loss factor must be increased by roughly a factor of 5 during the outsurge portion of any transient containing a strong outsurge.

Finally, DPC's model indicates an inability to accurately model reverse flow through stopped RCPs during coastdown of the other RCPs.

2.2 McGuire and Catawba Plant Model

Since these are not identical Westinghouse 4-loop plants, DPC developed different RETRAN models starting from the same basic model. Modifications were made in each analysis to better model the specific plant introducing some design differences between McGuire and Catawba plants and unit-dependent differences between two units of McGuire and Catawba. However, DPC assumed that the differences between the plants were small enough that model qualification through benchmark analysis of one should be considered to support the model developed for the other.

In addition, DPC developed two different models of McGuire and Catawba Plants: (1) one-loop plant model and (2) two-loop model. The one-loop model is to be used when all four loops are expected to behave similarly so that there is no asymmetrical condition. The two-loop model is to be used when asymmetric conditions are expected in the plant during the transient, thus one affected loop was modeled separately while the other three loops are lumped together. Although no details were presented, DPC also developed a three-loop model using the same basic approach.

A detailed description of the plant nodalization and models selected for use in transient analysis was presented in Chapter 3 of the topical report. The steam generator model contained a multiple number of volumes in the secondary side. DPC selected the RETRAN internal model for all volumes after an extensive series of parametric studies (Refs. 13 and 14). The mixture level prediction is made as a function of differential pressure across the location of pressure taps rather than to attempt to compute the mixture level. DPC is aware of the inability of its model to compute a mixture level.

The pressurizer is represented by a non-equilibrium volume.

RETRAN control systems were developed and used extensively by DPC to specify transient boundary conditions, such as automatic plant actions and operator actions as well as control actions by modulating valves, changing fill rates or reactivity and simulation of trip actuation. The control system is also used to compute the steam generator level by emulating the plant measurement devise by computing DP across the locations of the SG pressure taps. It is also used to convert a predicted mixture levels in the pressurizer into an indicated level and incorporating time delays into the predicted RCS loop temperatures to convert to the indicated temperatures. In all cases, DPC attempted to simulate the actual plant measuring devices.

2.2.1 McGuire/Catawba RETRAN Model Qualification

Although in general DPC chose to demonstrate the adequacy of the base plant

model for McGuire/Catawba (M/C) plants through comparison of RETRAN analyses to available plant data, in response to NRC questions, DPC provided (Refs. 13 and 14) details of sensitivity studies performed to assess adequacy of its M/C nodalization, in particular its steam generator model, and certain model and input selections. DPC provided thorough analyses of parametric sensitivity studies. The M/C plant RETRAN model was found to be acceptable not only in application for best-estimate analyses but also for licensing type analyses subject to the limitations set forth in the SERs on the topical reports DPC-NE-3001 and DPC-NE-3002.

This evaluation is based upon a review of the ability of the base model to benchmark startup test data and several operational transient data in a wide range of plant conditions.

2.2.2 Benchmark Analyses

For this objective, DPC performed benchmark analyses of 8 tests and plant transients, of which two were from the Catawba plants and the rest were from the McGuire plants.

The one-loop model was used for (1) Loss of Main Feedwater from 30% Power, (2) Steam Line PORV Failures, (3) Loss of Offsite Power and (4) Turbine Trip Test at 68 % Power.

The two-loop model was used for (1) loss of Main Feedwater to One Steam Generator and (2) Reactor Coolant Pump Trip at 89% Power.

For the reactor coolant pump flow coastdown tests, DPC simplified the base RETRAN models to only model the primary loop without any thermal modeling. The one-loop model simulated the four pump coastdown while two-, three- and four-loop models were also used to modeling consistency. The three-loop model was used for other combination of pump configuration during the tests.

The natural circulation test was simulated by use of two plant models: the one-loop as the base case and the three-loop model for the case with

sequential isolation of SGs.

2.2.2.1 Loss of Main Feedwater from 30% Full Power

In the benchmark analysis of the loss of main feedwater event from 30% full power, DPC used the one-loop McGuire Unit 2 RETRAN model.

For this analysis, DPC developed control systems: to match the pre-trip steam line pressure data, to match the post-trip steam line pressure response, and to regulate PZR spray flow. These as well as MFW and AFW flows were used as boundary conditions. Charging and letdown flows were not modeled.

The RETRAN results and plant data agreed reasonably well between 0 and 150 seconds. After roughly 150 seconds, DPC postulated two contributors to the discrepancies in pressurizer parameters between RETRAN results and plant data: (1) pressurizer backup heaters were predicted to de-energized by RETRAN but did not actually shut off, and (2) the absence of modeling of the charging and letdown system in the RETRAN model. The belief that charging and letdown actually had been activated at the plant was supported by a hand calculation by DPC.

The RETRAN control system used to compute the steam generator NR level was based upon the DP measurements between two pressure taps, and therefore was dependent upon nodalization. Anomalous behavior originating from the pressure and mass computation in the steam generator secondary (related to "pancaking") had little overall impact upon the global transient behavior in this analysis.

2.2.2.2 Loss of Main Feedwater to One Steam Generator

A two-loop McGuire Unit 2 model was used for this analysis. Boundary conditions used were MFW and AFW flows. Charging and letdown flows were not modeled.

The prediction of PZR pressure diverged (overpredicted) from the data. DPC

explained the early portion of the overprediction as being due to the absence of modeling of steam-liquid heat transfer in the PZR, and the latter portion as being due to an error in modeling the PZR heaters.

Imprecision in modeling the loop A steam line PORV and the condenser dump valves was postulated by DPC to be the source of the failure of the RETRAN computation to model the spikes in steam line pressure.

2.2.2.3 Steam Line PORV Failure

This was an event initiated by a test conducted at the Catawba Nuclear Station Unit 2 which went beyond the intended range due to an operator error. The plant was operating at 24% power when the test was initiated. When the control breakers were tripped, all four steam line PORVs opened and remained open for six minutes.

DPC specified AFW flow, auxiliary steam loads, charging and letdown flows and safety injection flow as boundary conditions. The steam line PORV junction area was adjusted to match the steam line depressurization rate.

Using the one-loop Catawba Unit 2 model, DPC obtained good agreement with the plant data for the key plant parameters presented in the topical report with the exception of the SG level. DPC stated that the underprediction was due to low initial SG inventory and uncertainty in AFW.

2.2.2.4 Reactor Coolant Pump Coastdown Tests

The reactor coolant (RCP) pump coastdown tests were performed as part of the pre-critical startup testing under isothermal conditions with the reactor subcritical. These tests serve to confirm the flow coastdown characteristics.

For this benchmark analyses, DPC used the model consisting of only the primary loops without any thermal modeling. In addition, both one- and three-loop models were used after unit specific models were developed to

determine impact of any unit design dependent differences.

RETRAN predicted parameters were comparable to those obtained during the tests. These results validated DPC's RCP model to simulate RCP coastdown characteristics over the range of flows indicated in the report.

2.2.2.5 Natural Circulation Testing

Two types of natural circulation tests were submitted to support natural circulation modeling for McGuire and Catawba: steady-state natural circulation tests conducted at 1% and 3% full power at both plants, although there is some uncertainty in the core power; and a test conducted at McGuire to evaluate the plant response to isolating two SGs in sequence after achieving a stable natural circulation condition with the reactor critical at approximately 1% power. In this latter test, SGs were isolated by closing the MSIV, isolating feedwater, and isolating blowdown.

For the steady-state test, the one-loop McGuire Unit 1 model was used for analysis while a three-loop McGuire Unit 1 model was used for the natural circulation with SG isolation test simulation.

The computed trend was in the same direction as the test data in the steady-state natural circulation tests; however, no further conclusion can be drawn from this comparison due to plant power level uncertainties.

In the natural circulation with SG isolation test simulation, the predicted and test data did not agree well.

The differences were attributed to inaccurate modeling of reactor power, overprediction primary-to-secondary heat transfer, potential steam leaks and ambient cooling from isolated SGs.

2.2.2.6 Reactor Coolant Pump Trip from 89% Full Power

An RCP trip from 89% full power occurred at McGuire Unit 1 when the DPC "C"

bus feeder breaker opened. Because of the asymmetric nature of the event, DPC used the two-loop McGuire Unit 1 RETRAN model for analysis. The RETRAN simulation was performed by adjusting the RCS flow to match core delta T. The steam line pressure data was input by DPC as a boundary condition during the simulation to better match the actual plant performance, since plant valve position data was unavailable and using a best-estimate resulted in discrepant results.

RETRAN predicted plant parameters were comparable to plant data. The difficulty in matching the steam generator level in the first 40 seconds of the transient was again attributed by DPC to non-physical mass redistributions caused by the RETRAN modeling of two phase flows in the steam generator secondary side.

2.2.2.7 Loss of Offsite Power

Plant data were obtained during the loss of offsite power event initiated by a spurious high power range flux rate which tripped the reactor at 100% full power operation. A one-loop plant model was used. The RCS flow was specified to match delta T.

Plant steam line pressure, MFW, AFW, charging and letdown flows, and status of PZR heater banks were specified as boundary conditions.

DPC's computation of the loss of offsite power event resulted in an underprediction in the pressurizer pressure beginning at about 100 seconds reaching a 150 psi underprediction by roughly 400 seconds and remaining there for the balance of the computation. The loop delta T's were similarly underpredicted by roughly 20%, with Thot being underpredicted by approximately 10 degrees and Tcold being matched. DPC attributed these differences to underprediction of loop hydraulic losses at low flow.

2.2.2.8 Turbine Trip Test from 68% Full Power

A Turbine Trip Test from 68% Full Power was conducted as an Operational

Transient Without Reactor Trip at the Catawba Nuclear Station Unit 1. This test is performed to demonstrate the effectiveness of plant control systems to stabilize the plant without tripping the reactor. In the one-loop Catawba Unit 1 RETRAN model, DPC stated that it built in detailed modeling of the pressurizer pressure controller and the plant control systems including operator actions.

The RCS flow was specified in the simulation to match delta T. The boundary conditions include the main feedwater flow rate and the reference T-ave as a function of time.

The results indicated only general trend agreement, since the power was inaccurately simulated after approximately 90 seconds and therefore the other plant parameters were not well matched.

2.2.3 Summary

DPC was able to get better agreement in the McGuire/Catawba benchmark analyses than in Oconee analysis, largely because the primary-to-secondary heat transfer was less dependent upon the secondary side modeling because the SG tubes remain covered in most transients.

However, as before, the RETRAN results consistently showed inaccurate modeling of natural circulation flow although this may be caused by uncertainties associated with test data.

In most instances when the measured data and RETRAN predicted results did not agree, the sources of differences were generally attributed by DPC to be due to inaccuracies or lack of sufficient details in the measured data.

Finally, the controller model of the steam generator level continuously gave spurious indications due to the manner in which RETRAN computed the steam generator pressures in the stacked volumes.

3.0 CONCLUSIONS

DPC topical reports DPC-NE-3000 and its supporting documents, including the DPC responses to NRC questions, were reviewed.

Based upon the submitted materials and through analysis of plant transient behavior using RETRAN, DPC has exhibited a high degree of staff technical competence, both in knowledge of the plants themselves and in understanding plant transient behavior. In addition DPC staff has demonstrated an excellent analytical knowledge of the code and code models. Furthermore, DPC staff has demonstrated sophistication in its use of the RETRAN control systems.

DPC's RETRAN models for the McGuire/Catawba nuclear power plants are generally acceptable, and acceptability extends to application to the licensing type analyses provided that analyses contain adequate conservatisms to produce acceptable results, and subject to the limitations set forth in the SERs on DPC-NE-3001 and 3002, and provided further that the following condition is satisfied:

> With respect to modeling under-cooling transients caused by loss of or reduction in feedwater flow, use of the steam generator modeling is acceptable for all transients in that category subject to the following condition:
>
> (1) if the affected steam generator(s) is/(are) relied upon for heat removal, the secondary side inventory never decreases below an amount which, if collapsed to zero void fraction, would cover enough tube height to remove decay heat.

DPC's RETRAN models for the Oconee plants require further justification of the steam generator model before it can be used in either best-estimate or licensing type analyses and in particular DPC must demonstrate that;

(1) its steam generator secondary side modeling produces conservative results for each such transient;

(2) its nodalization for the reactor vessel is appropriate for the transient to be analyzed and conservative;

(3) its selection of RETRAN internal models and correlations is conservative; and

(4) its RETRAN control systems are accurate and conservative.

4.0 REFERENCES (Part 1 - RETRAN)

1. "Thermal-Hydraulic Transient Analysis Methodology," DPC-NE-3000, July 1987.

2. Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (Generic Letter No. 83-11), USNRC, February 8, 1983.

3. "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM Revision 2, EPRI, November 1984.

4. "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI NP-2511-CCM Revision 2, EPRI, July 1985.

5. Letter, C.O. Thomas (NRC) to T.W. Schnatz (UGRA), September 4, 1984, (Transmittal of RETRAN-02 Safety Evaluation Report).

6. Letter, C.E. Rossi (NRC) to J.A. Blaisdell (UGRA), May 1, 1986, (Transmittal of VIPRE-01 Safety Evaluation Report).

7. Letter from H.B. Tucker (DPC) to USNRC, Attachment "DPC-NE-3000 Revision 1," May 11, 1989.

8. Letter from H.B. Tucker (DPC) to USNRC, Attachment "Duke Power Responses to NRC Questions Dated April 7, 1989 Regarding DPC-NE-3000," June 15, 1989.

9. Letter from H.B. Tucker (DPC) to USNRC, "Responses to Questions Regarding Differences Between Duke Topical Reports DPC-NE-2003 and DPC-NE-3000," June 19, 1989.

10. Letter from H.B. Tucker (DPC) to USNRC, "Duke Power Response to NRC Questions Regarding Steam Generator Heat Transfer Modeling with the RETRAN Code," August 9, 1989.

11. Letter from H.B. Tucker (DPC) to USNRC, Attachment 1 "Response to NRC

Questions on the McGuire/Catawba Sections of DPC-NE-3000" and Attachment 2 "Revisions to Section 4 of DPC-NE-3000," September 13, 1989.

12. Letter from M.S. Tuckman (DPC) to USNRC, "Supplemental Information to Assist in Review of Topical Reports DPC-NE-3000 and DPC-NE-2004," August 29, 1991.

13. Letter from H.B. Tucker (DPC) to USNRC, "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

14. Letter from H.B. Tucker (DPC) to USNRC, "Final Response to Questions Regarding the Topical Reports Associated with the M1C8 Reload Package," November 5, 1991.

TECHNICAL EVALUATION OF THE THERMAL-HYDRAULIC TRANSIENT ANALYSIS METHODOLOGY TOPICAL REPORT DPC-NE-3000 FOR THE DUKE POWER COMPANY OCONEE, MCGUIRE AND CATAWBA NUCLEAR STATIONS Part 2

1.0 INTRODUCTION

DPC-NE-3000, dated July 1987 (Ref. 1), documents results of a series of studies performed by Duke Power Company (DPC) to support the development of thermal-hydraulic transient analysis methods. Part 1 of this Technical Evaluation Report (TER) documents evaluation, in accordance with the RETRAN Safety Evaluation Report (SER) (Ref. 2), of DPC's use of the RETRAN computer code (Ref. 3) and the acceptability of the DPC RETRAN models for analysis of Oconee and McGuire/Catawba Nuclear Stations. Part 2 contains evaluation, in accordance with the VIPRE SER (Ref. 4), of DPC's intended use of the VIPRE-01 computer code (Ref. 5) in transient DNBR calculation and its conformity of the DPC submittals to the VIPRE SER requirements.

During the course of review of DPC-NE-3000, the chapter presenting Oconee VIPRE models was replaced in its entirety with Revision 1 of the topical report, at which time the McGuire/Catawba VIPRE model qualification chapter was added to the subject topical report as part of Chapter 3 (Ref. 6). Therefore, this review is based upon review of Revision 1 to DPC-NE-3000.

Two different VIPRE models for the core thermal-hydraulic analysis have been developed by DPC for use in steady-state, documented in DPC-NE-2003 and DPC-NE-2004, and transient applications for both types of plants (Refs. 7 and 8). Transient application of VIPRE-01 for both Oconee and McGuire/Catawba are

reviewed herein. DPC documented the differences between the models used for steady state and those used for transient applications (Refs. 6 and 9); the steady-state model is used in support of core reload analysis and the transient model is used for FSAR Chapter 15 type analysis. For these two applications, DPC uses different assumptions, nodalizations, thermal-hydraulic models and correlations, and other input data selections. Therefore, it was necessary for DPC to fully justify its intended use of VIPRE in transient applications.

The DPC submittal contains DPC's geometric representation of the core, its selection of thermal-hydraulic models and correlations, and a description of the methodology used for FSAR Chapter 15-type licensing transient analysis. Although DPC's basic methodology and conservative assumptions to be used for FSAR Chapter 15-type analysis are the same in both Oconee and McGuire/Catawba plants, evaluation is presented here separately for each type of plants.

2.0 EVALUATION

2.1 VIPRE Model Description

VIPRE-01 has been previously reviewed and approved for application to pressurized water reactor (PWR) plants in steady-state and transient analyses with heat transfer regimes up to critical heat flux. The VIPRE-01 SER includes conditions requiring each user to document and submit to the NRC for approval its procedure for using VIPRE-01 and to provide justification for its specific modeling assumptions, choice of particular two-phase flow models and correlations, heat transfer correlations, CHF correlation and DNBR limit, input values of plant specific data such as turbulent mixing coefficient and grid loss coefficient including defaults. This topical report was prepared to address these issues.

2.2 Oconee Core Analysis

The Oconee reactor core consists of 177 BAW Mark-BZ fuel assemblies. Each fuel assembly is a 15 x 15 array containing 208 fuel rods, 16 control rod

guide tubes, and one incore instrument guide tube.

2.2.1 Core Nodalization

In its sensitivity studies, DPC used the final set of thermal-hydraulic models and correlations which DPC intends to use in future licensing analysis.

2.2.1.1 Radial Noding Sensitivity

Since the VIPRE-01 code performs the thermal-hydraulic calculations simultaneously for all subchannels (a single-pass approach) and permits flexibility in selection of channel sizes and shapes, a sensitivity study was performed to determine the sensitivity of predicted DNBR to the subchannel model sizes. DPC intends to use the symmetrical case for the normal steady-state operation and most of the transients.

For asymmetrical cases, DPC will submit for NRC approval descriptions and justification of modeling of asymmetrical transients in separate submittals for NRC approval.

To assess nodalization sensitivity, DPC selected three different numbers of channels for core models using the same thermal-hydraulic correlations and models which DPC intends to use in future licensing analysis.

Sensitivity to the core model size was studied by comparing the results obtained with the coarse and fine size channel models. The coarse channel model was found to yield comparable MDNBRs as those obtained with the fine model. We therefore find DPC's use of the coarse channel model acceptable for Oconee thermal-hydraulic analysis.

2.2.1.2 Axial Noding Sensitivity

Using the coarse core model, three parametric calculations, each with BWC CHF correlation, were performed to assess sensitivity to the axial noding sizes.

The axial node lengths were selected by dividing the axial length into equal length of nodes. Two smaller node sizes correspond to the range of the code developer's recommended values. The results indicated that the mid-sized axial noding produced nearly identical MDNBR with those using the fine noding. We, therefore, find that use of the mid-sized uniform length axial nodes (Ref. 10) is acceptable for Oconee thermal-hydraulic analyses.

2.2.2 VIPRE-02 Input Data

DPC's approach to generation of input to the VIPRE-01 code was reviewed for acceptability. No review was conducted of the input data in comparison to the actual physical geometry.

2.2.2.1 Active Fuel Length

Since power is distributed over the length of the active fuel, a shorter aggregate fuel length yields higher power density, causing greater heat flux and is therefore conservative. DPC's choice for the active fuel length is conservative and acceptable. When a different assumption is used, DPC will justify its conservatism.

2.2.2.2 Spacer Grid Form Coefficients

Pressure losses across the spacer grids impact the axial pressure distribution and therefore the axial location of DNB. The spacer grid form loss coefficients were obtained from tests conducted by B&W. To determine the individual subchannel form loss coefficient, DPC stated that the vendor used its computer code, GRIL. The input data to the GRIL code are the individual subchannel geometry, drag areas and coefficients, and the coolant information. From this input, the code calculates individual subchannel loss coefficients, an overall grid loss coefficient and subchannel velocities based on single-phase flow input data by a iterative process. The calculated overall grid loss coefficient is matched with the measured value by adjusting the velocity field in the subchannel until consistency between the measured and predicted values is achieved. DPC has stated that the calculated

velocity profiles were compared by the vendor with the experimental data and showed good agreement (Ref. 11).

2.2.2.3 Core Bypass Flow

DNB is influenced by the aggregate flow rate past the location being examined, and therefore by the core bypass flow. Since the bypass flow depends on the number of control rod and burnable poison rod assemblies in the core, this is a cycle dependent parameter. Therefore, the core bypass flow data used in the analysis should be based on a bounding value or on cycle specific data. For the purpose of this submittal, the value DPC used is acceptable.

2.2.2.4 Inlet Flow Distribution

CHF is decreased and the probability of DNB is enhanced if flowrate is reduced due to a flow maldistribution. The use of 5% inlet flow maldistribution to the hot assembly with all four reactor coolant pumps operating was previously approved for Oconee FSAR analysis.

For operation with less than four reactor coolant pumps operating, more restrictive flow reduction factors are applied.

2.2.2.5 Flow Area Reduction Factor

DPC reduced the hot subchannel flow area by 2% to account for variations in as-built subchannel coolant flow area.

2.2.2.6 Radial Power Distribution

The reference design power distribution was developed using a radial-local hot pin peak which has been previously approved for Oconee FSAR analysis. DPC will submit for NRC approval a description and justification of applicability of its findings involving an asymmetrical radial power distribution.

2.2.2.7 Axial Power Distribution

The axial power shape used in the symmetric radial power distribution transients was a cosine shape with a peaking factor consistent with the current practice. DPC will justified any specific power shape for use on a case-by-case basis.

Prior to increasing the axial peaking factor, DPC will perform a complete evaluation of all potential safety concerns and submit it to the NRC for approval.

2.2.2.8 Hot Channel Factors

The power factor, F_q, used to account for variations in average pin power caused by differences in the fuel loading per rod is 1.0107 and is statistically determined from uncertainties associated with fuel.

DPC stated that their use of the local heat flux factor, F_q'', used to account for the uncertainty in the manufacturing tolerances, is consistent with the current application of the NRC approved methodology described in the DPC topical report NFS-1002.

2.2.2.9 Fuel Pin Conduction Model

DPC stated that for most of the transient analyses, the RETRAN heat flux boundary condition is used instead of the VIPRE-01 fuel pin conduction model. DPC further stated that for transient analyses in which the fuel enthalpy or cladding temperature is the protective criteria, the VIPRE-01 fuel pin conduction model may be used. DPC stated that evaluation of an appropriate approach would be made on a case-by-case basis for each analysis. DPC will provide justification for its selection of the conduction model.

2.2.2.10 Numerical Solution Technique

For the Oconee analyses presented in the submittal, DPC used the iterative solution method. However, should convergence be a problem, DPC will use the RECIRC solution method for Oconee FSAR type transient analyses.

2.2.3 VIPRE-01 Correlations

VIPRE-01 requires empirical correlations for the following models:

a. turbulent mixing
b. two-phase flow correlations (subcooled and saturated void, and void-quality relation)
c. critical heat flux

2.2.3.1 Turbulent Mixing

The lateral momentum equation requires two parameters: a turbulent momentum factor and a turbulent mixing coefficient.

The turbulent momentum factor (FTM) describes the efficiency of the momentum mixing: 0.0 indicating that crossflow mixes enthalpy only; 1.0 indicating that crossflow mixes enthalpy and momentum at the same strength. DPC selected values for both of these parameters are conservative.

2.2.3.2 Subcooled Void, Bulk Void and Two-Phase Flow Correlations

For subcooled and bulk void correlations, a sensitivity study using five different combinations of three subcooled and five bulk void correlations was performed using four cases varying only one boundary condition at a time. In all cases, the Columbia/EPRI two-phase friction multiplier was used. The results indicated that the DPC selected set of correlations predicted acceptably conservatively DNBRs relative to other combinations of correlations. DPC intends to use this combination in Oconee FSAR Chapter 15

analysis.

2.2.3.3 Critical Heat Flux Correlation

The B&W BWC CHF correlation using the LYNX-2 computer code has been reviewed and approved by the NRC for licensing analysis of BAW Mark-BZ fuel with Zircaloy grids with the design MDNBR limit of 1.18. The use of BWC correlation with VIPRE-01 has been also reviewed and approved by the NRC with the design MDNBR limit of 1.18 (Ref. 11).

Other correlations that may be utilized to cover other ranges of pressures are: W-3S (less than 1600 psia), MacBeth and Bowring (WSC-2) for low pressure and low flow conditions. DPC will provide justification when applying these correlation in future analyses.

2.2.4 Summary

We find that the subject topical report, together with DPC responses, contains sufficient information to satisfy the VIPRE-01 SER requirement that each VIPRE-01 user submit a document describing proposed use, sources of input variables, and selection and justification of correlations as it relates to use by DPC for FSAR Chapter 15 analyses.

For asymmetric transients, DPC intends to use other models not described in this submittal. Therefore, it is recommended that NRC approval be given for analysis of symmetric transients only.

In some instances, DPC selected default options since results are found to be insensitive to selection of parameters. In future licensing analyses, if changing any parameter results in less conservative prediction, DPC should submit justification of the change.

The B&W BWC CHF correlation with VIPRE-01 has been approved by the NRC with the design MDNBR limit of 1.18. DPC will provide justification as necessary when using other CHF correlation in future analyses.

Because the core bypass flow is cycle dependent, DPC will demonstrate, in future application, that its use of a particular core bypass flowrate is conservative.

Acceptability of DPC Oconee VIPRE-01 model is based upon selection of models/correlations supported by the sensitivity study results submitted. Should DPC change any of these items, DPC will submit justification for the change to the NRC for approval.

2.3 McGuire and Catawba Core Analysis

McGuire and Catawba Nuclear Stations each have two Westinghouse units and are assumed identical for the purpose of core thermal-hydraulic calculations. The analyses presented in the submittals assume BAW Mark-BW fuel assemblies which are assumed to be mechanically and hydraulically compatible with Westinghouse standard and optimized 17x17 fuel.

2.3.1 Core Nodalization

DPC used the final set of thermal-hydraulic models and correlations in the nodalization sensitivity studies which DPC intends to use in future licensing analysis.

2.3.1.1 Radial Noding Sensitivity

A parametric study was performed to determine the sensitivity of predicted DNBR to the subchannel model sizes. The thermal-hydraulic calculations were performed for three different core subchannel models using steady-state and transient conditions. Four transient cases were analyzed varying one boundary condition while keeping the others fixed.

The coarse channel model was found to yield acceptably conservative MDNBRs. Therefore, DPC intends to used the coarse channel model for FSAR type transient analyses for the McGuire and Catawba Nuclear Station.

However, for asymmetrical transients, DPC will submit a description and justification of modeling of asymmetrical transients coarse channel model in separate submittals.

2.3.1.2 Axial Noding Sensitivity

A sensitivity analysis for axial node length was performed with the coarse core channel model using three different sets of equal length axial nodes. Two finer node sizes correspond to the range of the code developer's recommended values. The results indicated that the mid size noding is adequately conservative. Therefore, we find that use of the mid-size uniform length axial nodes (Ref. 10) is acceptable for McGuire and Catawba thermal-hydraulic analyses.

2.3.2 VIPRE-01 Input Data

DPC's approach to generation of input to the VIPRE-01 code was reviewed for acceptability. No review was conducted of the input data in comparison to the actual physical geometry.

2.3.2.1 Active Fuel Length

For B&W's low densification fuel, the amount of fuel densification is off-set by the fuel thermal expansion. Therefore, it is more conservative to use the cold nominal active fuel length for calculation and this is acceptable.

2.3.2.2 Spacer Grid Form Coefficients

The same procedure used to determined these coefficients for Oconee core analysis was used for McGuire/Catawba grid form coefficients.

2.3.2.3 Core Bypass Flow

Since the bypass flow depends on the number of control rod and burnable

poison rod assemblies in the core, this is a cycle dependent parameter. Therefore, the core bypass flow data used in the analysis should be based on a bounding value or on justified cycle specific data. For the purpose of this submittal, the value DPC used is acceptable.

2.3.2.5 Inlet Flow Distribution

CHF is decreased and the probability of DNB is enhanced if flowrate is reduced due to a flow maldistribution. The use of 5% inlet flow maldistribution to the hot assembly with all four reactor coolant pumps operating yielded slightly more conservative results than a uniform inlet flow distribution.

For operation with less than four reactor coolant pumps operating, more restrictive flow reduction factors are applied.

2.3.2.6 Flow Area Reduction Factor

DPC reduced the hot subchannel flow area by 2% to account for variations in as-built subchannel coolant flow area.

2.3.2.7 Radial Power Distribution

The assembly and pin radial power distributions were selected assuming maximum peaking factors. A shape assumed for the assembly power distribution is designed to minimize flow redistribution. The same rational is used for the pin radial power distribution.

2.3.2.8 Axial Power Distribution

The axial power shape was selected to yield DNBR margin in the Chapter 15 transients and peaking margin compared to cycle specific power distributions. Use of this power shape and the radial power distribution is to use a design power distribution to ensure DNB protection.

2.3.2.9 Hot Channel Factor

The hot channel factor $F_{\Delta H}^{E}$ used for the McGuire/Catawba analysis is 1.03 and is the allowance on enthalpy rise to account for manufacturing tolerances. The value was determined by B&W.

2.3.2.10 Numerical Solution Technique

For the McGuire/Catawba analyses presented in the submittal, DPC used the RECIRC solution method. DPC will use the RECIRC solution method in future FSAR-type transient analyses (Ref. 10).

2.3.3 VIPRE-01 Correlations

VIPRE-01 requires empirical correlations for the following models:

a. turbulent mixing
b. two-phase flow correlations (subcooled and saturated void, and void-quality relation)
c. critical heat flux

2.3.3.1 Turbulent Mixing

The lateral momentum equation requires two parameters: a turbulent momentum factor and a turbulent mixing coefficient.

The turbulent momentum factor (FTM) describes the efficiency of the momentum mixing: 0.0 indicating that crossflow mixes enthalpy only; 1.0 indicating that crossflow mixes enthalpy and momentum at the same strength. DPC selected a conservative value for FTM.

Since the turbulent mixing coefficient determines the flow mixing rate, it is an important parameter. Based upon tests using a 5x5 heated bundle conducted by B&W, where the subchannel exit temperatures were measured, a mixing

coefficient was conservatively determined for B&W Mark-BW fuel which is proportional to the turbulence intensity (Ref. 10). For conservatism, DPC used a number smaller than the B&W determined coefficient and this reduced value will be used in the McGuire and Catawba core thermal-hydraulic analysis (Ref. 10).

2.3.3.2 Subcooled Void, Bulk Void and Two-Phase Flow Correlations

For subcooled and bulk void correlations, a sensitivity study using five different combinations of three subcooled void and five bulk void correlations was performed for steady-state and transient boundary conditions. The results indicated that the use of the DPC selected combination of correlations in conjunction with Columbia/EPRI two-phase friction multiplier predicted conservatively computed DNBR relative to other combinations of correlations. DPC intends to use this combination in McGuire and Catawba analysis.

This is consistent with the VIPRE-01 SER findings.

2.3.3.3 BWCMV Critical Heat Flux Correlation

Use of BWCMV CHF correlation with the LYNX2 code has been approved by the NRC for the DNBR limit of 1.21. Its use with VIPRE-01 has been also approved (Ref. 12).

2.3.4 Statistical Core Design Methodology

The DPC developed statistical core design methodology (SCD) statistically combines uncertainties associated with key parameters used in determination of the DNBR. Details of the methodology with respect to the steady-state application is documented in DPC-NE-2004. The transient application is performed in the same manner as described in that topical report.

During the review of DPC-NE-2004, in response to the NRC question, DPC provided results of sensitivity cases using models developed in DPC-NE-2004

and DPC-NE-3000. There were negligible differences between the predicted DNBRs (Refs. 13 and 14). Therefore, the SCD methodology developed in the DPC-NE-2004 is applicable in transient applications since the methodology allows enough margin in the DNBR limits to account for the small differences between two models. However, the same conditions cited in the technical evaluation report for DPC-NE-2004 are applicable to use of the SCD methodology in transient applications.

2.3.5 Summary

For asymmetric transients, DPC intends to use other models not described in this submittal. Therefore, it is recommended that NRC approval be given for use in analysis of symmetric transients only.

Because the core bypass flow is cycle dependent, DPC will demonstrate, in future application, that its use of a particular core bypass flowrate is conservative.

Acceptability of DPC M/C VIPRE-01 model is based upon selection of models/correlations supported by the sensitivity study results submitted. Therefore, whenever DPC changes any of these items documented in the topical report, DPC will submit justification for the change to the NRC for approval.

Furthermore, the use of the SCD methodology in transient application is acceptable provided that the range of applicability of the RSM is not violated. The conditions cited (Refs. 12 and 13) in the review of DPC-NE-2004 are applicable to transient application as well.

3.0 CONCLUSIONS

We find that the subject topical report, together with DPC responses, contains sufficient information to satisfy the VIPRE-01 SER requirement that each VIPRE-01 user submit a document describing proposed use, sources of input variables, and selection and justification of correlations as it relates to use by DPC for FSAR Chapter 15 analyses.

and DPC-NE-3000. There were negligible differences between the predicted DNBRs (Refs. 13 and 14). Therefore, the SCD methodology developed in the DPC-NE-2004 is applicable in transient applications since the methodology allows enough margin in the DNBR limits to account for the small differences between two models. However, the same conditions cited in the technical evaluation report for DPC-NE-2004 are applicable to use of the SCD methodology in transient applications.

2.3.5 Summary

For asymmetric transients, DPC intends to use other models not described in this submittal. Therefore, it is recommended that NRC approval be given for use in analysis of symmetric transients only.

Because the core bypass flow is cycle dependent, DPC will demonstrate, in future application, that its use of a particular core bypass flowrate is conservative.

Acceptability of DPC M/C VIPRE-01 model is based upon selection of models/correlations supported by the sensitivity study results submitted. Therefore, whenever DPC changes any of these items documented in the topical report, DPC will submit justification for the change to the NRC for approval.

Furthermore, the use of the SCD methodology in transient application is acceptable provided that the range of applicability of the RSM is not violated. The conditions cited (Refs. 12 and 13) in the review of DPC-NE-2004 are applicable to transient application as well.

3.0 CONCLUSIONS

We find that the subject topical report, together with DPC responses, contains sufficient information to satisfy the VIPRE-01 SER requirement that each VIPRE-01 user submit a document describing proposed use, sources of input variables, and selection and justification of correlations as it relates to use by DPC for FSAR Chapter 15 analyses.

We further find that the manner in which the code is to be used for such analyses, selection of nodalization, models, and correlations provides, except as listed below, adequate assurances of conservative results and is therefore acceptable. Furthermore, the use of the DPC developed statistical core design methodology as approved in the Technical Evaluation Report on DPC-NE-2004 (Ref. 12) is approved for the transient application subject to the same conditions.

The following items are limitations regarding VIPRE-01 application presented in DPC-NE-3000 and its supplemental materials:

(1) Determination of acceptability is based upon review of selection of models/correlations for symmetric transients only. DPC submitted its asymmetric models in DPC-NE-3001 for NRC review and approval.

(2) When using the DPC developed SCD method, the licensee must satisfy the conditions set forth in Reference 12.

(3) Whenever DPC intends to use other CHF correlations, power distribution, fuel pin conduction model or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, DPC must submit its justification for NRC review and approval.

(4) Core bypass flow should be determined on cycle-by-cycle bases.

4.0 REFERENCES (Part 2 - VIPRE)

1. "Duke Power Company - The Thermal-Hydraulic Transient Analysis Methodology - Oconee Nuclear Station, McGuire Nuclear Station, Catawba Nuclear Station," DPC-NE-3000, July 1987.

2. Letter, C.O. Thomas (NRC) to T.W. Schnatz (UGRA), September 4, 1984, (Transmittal of RETRAN-02 Safety Evaluation Report).

3. "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM Revision 2, EPRI, November

1984.

4. Letter, C.E. Rossi (NRC) to J.A. Blaisdell (UGRA), May 1, 1986, (Transmittal of VIPRE-01 Safety Evaluation Report).

5. "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI NP-2511-CCM Revision 2, EPRI, July 1985.

6. Letter from H.B. Tucker (DPC) to USNRC, Attachment to "Thermal-Hydraulic Transient Analysis Methodology," May 11, 1989.

7. "Duke Power Company Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003, August 1988.

8. "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004, December 1988.

9. Letter from H.B. Tucker (DPC) to USNRC, "Response to Questions Regarding Differences Between Duke Topical Reports DPC-NE-2003 and DPC-NE-3000," June 19, 1989.

10. Letter from G.B. Swindlehurst (DPC) to USNRC, "Response to NRC Questions on DPC-NE-3000 Dated July 25, 1989," February 20, 1990.

11. "Technical Evaluation of the Core Thermal-Hydraulic Methodology Using VIPRE-01 Topical Report DPC-NE-2003 for the Duke Power Company Oconee Nuclear Station," ITS/NRC/89-2, July 3, 1989.

12. "Technical Evaluation of the Core Thermal-Hydraulic Methodology Using VIPRE-01 Topical Report DPC-NE-2004 for the Duke Power Company McGuire and Catawba Nuclear Stations," ITS/NRC/91-1, October 1991.

13. Letter from M.S. Tuckman (DPC) to USNRC, "Supplemental Information to Assist in Review of Topical Reports DPC-NE-3000 and DPC-NE-2004," August 29, 1991.

14. Letter from H.B. Tucker (DPC) to USNRC, "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.