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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT DPC-NE-3002, REVISION 1, "FSAR CHAPTER 15 SYSTEM

TRANSIENT ANALYSIS METHODOLOGY"

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369, 50-370

50-413, AND 50-414

1.0 INTRODUCTION

In Revision 1 of the Topical Report DPC-NE-3002 entitled "FSAR Chapter 15 System Transient Analysis Methodology" dated June 1994 (Reference 1), Duke Power Company (DPC) documented revisions reflecting changes due to (i) replacement of steam generators (SGs) for the McGuire Units 1 and 2 and Catawba Unit 1 stations, and (ii) methodology changes documented in DPC-NE-3000, Revision 1 (Reference 2). Corrections of typographical errors were also included. Additional information was provided in Reference 3.

The original Topical Reports DPC-NE-3000 (Reference 4) and DPC-NE-3002 (Reference 5) were reviewed and approved, subject to certain conditions (References 6 and 7).

Steamline break, rod ejection, dropped rod, and boron dilution events were not part of this review since these events are documented in DPC-NE-3001 (Reference 8), which has been reviewed and approved.

2.0 REPORT SUMMARY

DPC-NE-3002 (References 1 and 5) contains DPC's qualitative approach to performance of FSAR Chapter 15 type analysis for the McGuire and Catawba stations using methodology utilizing the RETRAN and VIPRE-01 computer codes described in DPC-NE-3000. It does not address justification, qualification, or demonstration of the approaches taken for the analysis. However, it does state the process DPC intends to use in determining initial and boundary conditions, transient assumptions and scenarios, and code models used in licensing applications for transient analysis.

Revision 1 of DPC-NE-3002 documents changes due to (i) the replacement of steam generators for McGuire Units 1 and 2 and Catawba Unit 1, and (ii) minor methodology changes presented in Revision 1 of DPC-NE-3000. Typographical errors were also corrected. Changes include analysis objectives, pressurizer

ENCLOSURE 1

and SG models, initial and boundary conditions, transient assumptions in terms of system component availability, and the use of statistical core design (SCD) methodology for DNBR computation.

3.0 EVALUATION

Acceptability of DPC's revisions of RETRAN models and assumptions for thermal-hydraulic calculations of FSAR Chapter 15 transient analysis of its McGuire/Catawba (M/C) plants is discussed below. Only those items which bear analytical or safety significance are discussed. Those items of a non-technical nature are not discussed.

3.1 Changes in McGuire and Catawba RETRAN Methodology

The RETRAN base models for M/C plants were qualified in DPC-NE-3000 and its Revision 1 for both best estimate and licensing-type, non-LOCA applications, subject to limitations described in the Safety Evaluation (SE) (References 6 and 9). Note that DPC's submittal of August 9, 1994, was identified then as Revision 3 to the DPC-NE-3000 report. That submittal has since been renumbered as Revision 1 to the original DPC-NE-3000 report by DPC's letter of September 12, 1995. The approved version of the original DPC-NE-3000 report was issued by DPC on August 8, 1995 (Reference 6). The NRC's SE for Revision 1 to the original DPC-NE-3000 report was issued on December 27, 1995 (Reference 9).

A change which impacted the documentation of DPC-NE-3002 was a change in the pressurizer modeling described in DPC-NE-3000, Revision 1. Thus, all sections that related to the previous modeling description were revised.

Also included in the revision of the RETRAN methodology is modeling of a Babcock & Wilcox (B&W) feedring steam generator (FSG) model. Details of the FSG nodalization and other associated changes due to SG replacement are presented in Reference 2. A significant impact is expected in the Feedwater System Pipe Break analysis results due to the design and location of the main feedwater nozzles, which is discussed in Section 3.3 of this evaluation.

3.2 SCD Transients

The core thermal-hydraulics for most of the transients considered in this Topical Report are analyzed using the DPC-developed and NRC-approved SCD methodology (Reference 10). For these transients, certain initial conditions used in the transient safety analysis are selected to be at nominal conditions, as qualitatively defined in the subject report, since the uncertainty associated with the initial conditions is accounted for in the SCD method.

Of those transients for which a DNBR computation is performed, there remain two transients (startup of an inactive reactor coolant pump at an incorrect temperature and steam line break) for which DNBR calculations are not performed using the SCD methodology. With this revision, DPC stated its intent to use the SCD methodology for reactor coolant pump (RCP) locked rotor, and steam generator tube rupture (SGTR).

Although in the locked rotor analysis the core flowrate is expected to fall below the minimum SCD parameter value, a statistical Monte Carlo propagation is performed to ensure that the statistical design limit remains acceptable. This approach was approved provided that the range of applicability of the critical heat flux (CHF) correlation is not violated. In the SGTR analysis, DPC stated that the range of applicability remained valid for SCD parameters.

3.3 Revised FSAR Transient Analysis

In this section those transient analyses, in which significant revisions are proposed, are highlighted and other revisions are briefly discussed.

3.3.1 Increase in Heat Removal by the Secondary System

Two transients in this category, which incorporated revisions, are (i) Feedwater System Malfunction Causing an Increase in Feedwater Flow, and (ii) Excessive Increase in Secondary Steam Flow. In both cases revisions are minor since the changes are primarily editorial reflecting methodology changes in DPC-NE-3000, Revision 1, and, therefore, are acceptable.

3.3.2 Decrease in Heat Removal by the Secondary System

All four transient analyses are affected by revisions in this category: (i) turbine trip, (ii) loss of offsite power, (iii) loss of normal feedwater, and (iv) feedwater system pipe break. Turbine trip is analyzed with respect to peak RCS and secondary side pressure, and the others are analyzed with respect to peak RCS pressure and DNB and/or long-term core coolability (potential for hot leg boiling).

3.3.2.1 Turbine Trip

A change in the assumption regarding the pressurizer (PZR) level control is introduced. DPC stated that the use of the level control in manual with the PZR heaters locked on will be worse with respect to high primary system pressure than the case when the PZR level control is in automatic. The staff concurs with this assumption.

3.3.2.2 Loss of Offsite Power

In addition to the potential challenges to peak RCS pressure, peak secondary side pressure, and DNB, DPC will analyze this transient with respect to long-term core cooling capability. Therefore, a new section was added to the report describing the analysis to demonstrate that natural circulation can be established after loss of offsite power. Transient assumptions are reasonable. With respect to the other transient objectives, changes introduced are benign.

3.3.2.3 Loss of Normal Feedwater

Assumptions regarding the initial SG inventory were revised. In the new approach, low instead of high SG level is assumed to maximize the secondary pressure. This is expected to cause an earlier reactor trip on the SG low-low level. The downward adjustment of the initial SG level introduces competing effects with respect to predicted peak primary and secondary pressures and DNBR.

This event is currently not a limiting transient in this category and is bounded by the turbine trip event. Therefore, its analysis is not required. However, DPC stated that an analysis may become necessary in the future due to hardware or methodology changes. In that event, DPC will need to perform sensitivity studies with respect to initial condition selections to ensure conservatism in the analysis.

3.3.2.4 Feedwater System Pipe Break

This transient is significantly impacted by implementation of the feedring steam generators, and requires three major assumption changes as a direct result of the design and location of the main feedwater nozzles. DPC's discussion of assumption changes and the impact of changes in transient results was reviewed and found to be reasonable.

The loss of offsite power coincident with reactor trip is assumed, resulting in RCP trip and delay in the startup of the diesel generators for safety injection. Early main steam isolation valve (MSIV) closure was determined to be conservative in terms of earlier faulted SG dryout. Thus, in the revised assumptions, MSIV closure occurs coincident with turbine trip, which occurs on loss of offsite power. DPC's approach to the analysis of this event is acceptable.

3.3.3 Decrease in Reactor Coolant System Flowrate

Three transients analyzed in this category are: (1) partial loss of forced reactor coolant (RC) flow, (2) complete loss of forced reactor coolant flow, and (3) reactor coolant pump locked rotor.

Revisions to both the complete and partial loss of forced RC flow are editorial changes and are acceptable.

3.3.3.1 Reactor Coolant Pump Locked Rotor

As stated in Section 3.2, DNBR for this event will be analyzed using the SCD methodology. Therefore, affected parameters are initially set to nominal values instead of assuming conservative values. DPC provided the explanation of the applicability of the SCD methodology for this transient (Reference 3) and the staff finds the explanation to be acceptable (see also Section 3.2).

DPC stated that cases with and without loss of offsite power coincident with the turbine trip will be analyzed.

As stated in the SE (Reference 7) for DPC-NE-3002 (Reference 5), the assumption of 120% of design pressure is not an acceptable limit. DPC is required to use 110% of design pressure, as stated in the previous revision.

3.3.4 Reactivity and Power Distribution Anomalies

DPC added the possibility of reactor trip on high pressurizer pressure in addition to the high neutron flux for completeness.

3.3.5 Increased Reactor Coolant Inventory

Inadvertent operation of ECCS during power operation is the only transient analyzed. Although DNB is a primary concern, since a potential for pressurizer overfill exists during this event, DPC added a new section to address that concern for PZR overfill leading to water relief through the PZR Safety Valves (PSVs). The acceptance criterion for this analysis is the minimum water relief temperature to assure PSV operability.

The Standard Review Plan suggests the use of full power unless a lower power can be justified. In Reference 3, DPC assumes zero power in this analysis for conservatism. This is because if overfill occurs at lower initial power, then the water relief temperature is more likely to be less than the acceptance criterion. Therefore, DPC selected the initial and boundary conditions in order to minimize relief temperature. The staff finds this approach to be reasonable and acceptable.

3.3.6 Decrease in Reactor Coolant Inventory

Inadvertent opening of a pressurizer safety or relief valve and steam generator tube rupture events are the two transients analyzed in this category. Proposed revisions to the inadvertent opening of a pressurizer safety or relief valve are editorial changes.

3.3.6.1 Steam Generator Tube Rupture

The steam generator tube rupture (SGTR) event was not part of the original review since the transient methodology documented in DPC-NE-3000, based on the use of the RETRAN computer code, was approved only for non-LOCA applications. This restriction regarding performance of SGTR analysis with RETRAN (Item vii of RETRAN SER (Reference 11)) applies to applications that encounter two-phase flow in the primary loop, which does occur in many SGTR scenarios.

In the limited review documented in Reference 12, DPC received approval for an SGTR analysis of the worst-case offsite dose scenario using RETRAN for Catawba Nuclear Station, Units 1 and 2. Justification was provided in a qualitative manner by DPC (Reference 13) on each of the items cited under restrictions and limitations on the use of RETRAN in its SE. There is assurance that the use of the code for that particular scenario was acceptable since DPC stated that two-phase flow was not encountered in the primary loop.

Although NRC approval was specific to Catawba Units 1 and 2, as considered in DPC-NE-3000, the Catawba and McGuire plants, for the purpose of analysis qualification, are interchangeable. Therefore, DPC stated that NRC approval of the SGTR analysis using RETRAN should be applicable to the McGuire plant analysis (Reference 3). The staff concurs with DPC's statement, so long as the scenario is essentially the same and no two-phase flow conditions are encountered in the RCS primary loops.

The DNBR will be computed using the SCD methodology (see Section 3.2).

4.0 CONCLUSIONS AND LIMITATIONS

Revision 1 to the DPC Topical Report DPC-NE-3002 and the DPC responses to NRC questions and other supporting documents cited in Section 5.0 were reviewed. Review of these documents focused upon evaluation of acceptability of the proposed changes and the perceived impact of these changes.

As stated earlier, steamline break, rod ejection, dropped rod, and boron dilution events were not part of this review.

Subject to the foregoing, DPC's proposed revision of its approach to FSAR Chapter 15 transient analysis, as documented in Revision 1 of DPC-NE-3002 and its supporting document, was found to be acceptable subject to the following limitations:

1. The acceptability of the use of DPC's approach to FSAR analysis is subject to the conditions of SEs on all aspects of transient analysis and methodologies (DPC-NE-3000, DPC-NE-3001, DPC-NE-3002, DPC-NE-2004, and DPC-NE-2005) as well the SEs on the RETRAN and VIPRE-01 computer codes.

2. There are scenarios in which an SGTR event may result in loss of subcooling and the consequent two-phase flow conditions in the primary system. In such instances, the use of RETRAN is not acceptable without a detailed review of the analysis.

3. In the future, if hardware or methodology changes, selection of limiting transients needs to be reconsidered, and DPC is required to perform sensitivity studies to identify the initial conditions in such a way to avoid conflict between transient objective, such as DNB and worst-case primary pressure.

4. It is emphasized that, when using the SCD methodology to determine DNBR, the range of applicability of the selected critical heat flux correlation must not be violated.

5. DPC's assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor is not acceptable; DPC is required to use 110% of design pressure for that limit.

Principal Contributor: L. Lois

Date: December 28, 1995

5.0 REFERENCES

1. Letter from M. S. Tuckman (DPC) to NRC, dated July 18, 1994, transmitting "FSAR Chapter 15 System Transient Analysis Methodology, DPC-NE-3002," Revision 1, June 1994.

2. Letter from M. S. Tuckman (DPC) to NRC, dated August 9, 1994, transmitting "Duke Power Company, The Thermal-Hydraulic Transient Analysis Methodology Oconee Nuclear Station, McGuire Nuclear Station, Catawba Nuclear Station," DPC-NE-3000-P, redesignated as Revision 1, August 1994.

3. Letter from M. S. Tuckman (DPC) to NRC, "Topical Report DPC-NE-3002, "FSAR Chapter 15 System Transient Analysis Methodology," Responses to NRC Questions," August 18, 1995.

4. Letter from H. B. Tucker (DPC) to NRC, dated September 29, 1987, transmitting "Duke Power Company, The Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, July 1987" for Oconee Nuclear Station, McGuire Nuclear Station, and Catawba Nuclear Station.

5. Letter from H. B. Tucker (DPC) to NRC, dated August 17, 1992 transmitting the approved version of the report "DPC-NE-3002-A, FSAR Chapter 15 System Transient Analysis Methodology."

6. Letter from T. A. Reed (NRC), to H. B. Tucker (DPC), dated November 15, 1991, transmitting "Safety Evaluation on Topical Report DPC-NE-3000, Thermal-Hydraulic Transient Analysis Methodology," as transmitted with the approved version of the report (DPC-NE-3000-PA) by M. S. Tuckman's letter of August 8, 1995.

7. Letter from T. A. Reed (NRC) to H. B. Tucker (DPC), dated November 15, 1991, "Safety Evaluation on Topical Report DPC-NE-3002, FSAR Chapter 15 System Transient Analysis Methodology," as transmitted with the approved version of the report (DPC-NE-3002-A) by H. B. Tucker's letter of August 17, 1992.

8. Letter from H. B. Tucker (DPC), to NRC, dated August 17, 1992, transmitting the approved version of "Multi-dimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," DPC-NE-3001PA, November 1991.

9. Letter from R. E. Martin (NRC), to M. S. Tuckman (DPC), dated December 27, 1995, transmitting "Safety Evaluation for Revision 1 to Topical Report DPC-NE-3000-P, Thermal-Hydraulic Transient Analysis Methodology."

10. Letter from G. M. Holahan (NRC) to H.B. Tucker (DPC), dated February 24, 1995, "Acceptance for Referencing of the Modified Licensing Topical Report, DPC-NE-2005P, Thermal-Hydraulic Statistical Core Design Methodology," as transmitted with M. S. Tuckman's letter of August 8, 1995, "Issuance of Approved Version of DPC-NE-2005P (DPC-NE-2005P-A)."

REFERENCES (continued)

11. Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," September 2, 1984.

12. Letter from R. E. Martin (NRC) to M. S. Tuckman (DPC), "Safety Evaluation for the Catawba Nuclear Station, Units 1 and 2, Steam Generator Tube Rupture Analysis," May 14, 1991.

13. Letter from H. B. Tucker (DPC) to NRC, dated December 7, 1987, "Catawba Nuclear Station Steam Generator Tube Rupture Analysis."

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TECHNICAL EVALUATION:
FSAR CHAPTER 15 SYSTEM TRANSIENT ANALYSIS METHODOLOGY
DPC-NE-3002 REVISION 1
FOR
DUKE POWER COMPANY

P.B. Abramson
H. Komoriya

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International Technical Services, Inc.
420 Lexington Avenue
New York, NY 10170

ENCLOSURE 2

TECHNICAL EVALUATION:
FSAR CHAPTER 15 SYSTEM TRANSIENT ANALYSIS METHODOLOGY
TOPICAL REPORT DPC-NE-3002 REVISION 1
FOR
DUKE POWER COMPANY
MCGUIRE AND CATAWBA NUCLEAR STATIONS

1.0 INTRODUCTION

In Revision 1 of the topical report entitled "FSAR Chapter 15 System Transient Analysis Methodology," DPC-NE-3002, dated June 1994 (Ref. 1), Duke Power Company (DPC) documented revisions reflecting changes due to (i) replacement of steam generators for the McGuire and Catawba Unit 1 stations and (ii) methodology changes documented in DPC-NE-3000 Rev. 3 (Ref. 2). Corrections of typographical errors were also included. Additional information was provided in Reference 3.

The original topical reports DPC-NE-3000 (Ref. 4) and DPC-NE-3002 (Ref. 5) were reviewed and approved, subject to certain conditions (Refs. 6 and 7).

DPC-NE-3002 (Refs. 1 and 5) contains DPC's qualitative approach to selection of initial and boundary conditions, transient assumptions and computer code models for use in performing transient analysis of FSAR Chapter 15 accidents for McGuire and Catawba Nuclear Stations. The report does not contain any justification, qualification or demonstration of selections.

Steam line break, rod ejection, dropped rod and boron dilution events were not part of this review since these events are documented in DPC-NE-3001 (Ref. 8) which has been reviewed and approved.

2.0 SUMMARY

DPC-NE-3002 contains DPC's qualitative approach to performance of FSAR Chapter 15-type analysis for the McGuire and Catawba stations using methodology utilizing the RETRAN and VIPRE-01 computer codes described in DPC-NE-3000. It does not address justification, qualification or demonstration of the approaches taken for analysis. However, it does state the process they intend to use in determining initial and boundary conditions, transient assumptions and scenarios and code models used in licensing-type transient analysis.

Revision 1 of DPC-NE-3002 documents changes due to (i) the replacement steam generators for McGuire and Catawba Unit 1 and (ii) minor methodology changes presented in Revision 3 of DPC-NE-3000. Typographical errors are also corrected. Changes include analysis objectives, pressurizer and SG models, initial and boundary conditions, transient assumptions in terms of system component availability, and the use of statistical core design methodology for DNBR computation.

3.0 EVALUATION

Acceptability of DPC's revisions of RETRAN models and assumptions for thermal-hydraulic calculations of FSAR Chapter 15 transient analysis of its McGuire/Catawba (M/C) plants is discussed below. Only those items which bear analytical or safety significance are discussed. Those items of a non-technical nature are not discussed.

3.1 Changes in McGuire and Catawba RETRAN Methodology

The RETRAN base models for M/C plants were qualified in DPC-NE-3000 and its Revision 3 for both best-estimate and licensing type non-LOCA applications, subject to limitations described in the SER and TER (Refs. 6 and 9).

A change which impacted the documentation of DPC-NE-3002 was a change in PZR modeling described in DPC-NE-3000 Rev. 3. Thus, all sections which related to previous modeling description were revised.

Also included in the revision of the RETRAN methodology is modeling of a B&W feedring steam generator (FSG) Model. Details of the FSG nodalization and other associated changes due to SG replacement are presented in Reference 2. A significant impact is expected in the Feedwater System Pipe Break analysis results due to the design and location of the main feedwater nozzles, which is discussed in Section 3.3. of this report.

3.2 SCD Transients

The core thermal-hydraulics for most of the transients considered in this topical report are analyzed using the DPC developed and NRC approved SCD methodology (Ref. 10). For these transients, certain initial conditions used in the transient safety analysis are selected to be at nominal conditions, as qualitatively defined in the subject report, since the uncertainty associated with the initial conditions is accounted for in the SCD method.

Of those transient for which a DNBR computation is performed, there remain two transients (startup of an inactive reactor coolant pump at an incorrect temperature and steam line break) for which DNBR calculations are not performed using the SCD methodology. With this revision, DPC stated its intent to use the SCD methodology for RCP Locked Rotor and SGTR.

Although in the Locked Rotor analysis the core flowrate is expected to fall below the minimum SCD parameter value, a statistical Monte Carlo propagation is performed to ensure that the statistical design limit remains acceptable. This approach was approved provided that the range of applicability of the critical heat flux (CHF) correlation is not violated.

In the SGTR analysis, DPC stated that the range of applicability remained valid for SCD parameters.

3.3 Revised FSAR Transient Analysis

In this section those transient analyses in which significant revisions are proposed are highlighted and other revisions are briefly discussed.

3.3.1 Increase in Heat Removal by the Secondary System

Two transients in this category which incorporated revisions are (i) Feedwater System Malfunction Causing an Increase in Feedwater Flow and (ii) Excessive Increase in Secondary Steam Flow. In both cases revisions are minor since the changes are primarily editorial reflecting methodology changes in DPC-NE-3000 Rev. 3 and therefore acceptable.

3.3.2 Decrease in Heat Removal by the Secondary System

All four transient analyses are affected by revisions in this category: (i) turbine trip, (ii) loss of offsite power, (iii) loss of normal feedwater, and (iv) feedwater system pipe break. Turbine trip is analyzed with respect to peak RCS and secondary side pressure, and the others are analyzed with respect to peak RCS pressure and DNB and/or long term core coolability (potential for hot leg boiling).

3.3.2.1 Turbine Trip

A change in the assumption regarding the PZR level control is introduced. DPC stated that the use of the level control in manual with the PZR heaters locked on will be worst in order to elevate the primary pressure to a higher value than is obtained when the PZR level control in automatic. We concur.

3.3.2.2. Loss of Offsite Power

In addition to the potential challenges to peak RCS pressure, peak secondary side pressure and DNB, DPC will analyze this transient with respect to long-term core cooling capability. Therefore, a new section was added to the report describing the analysis to demonstrate that natural circulation can be established after loss of offsite power. Transient assumptions are reasonable. With respect to the other transient objectives, changes introduced are benign.

3.3.2.3 Loss of Normal Feedwater

Assumptions regarding the initial SG inventory were revised. In the new approach, low instead of high SG level is assumed, to maximize the secondary pressure. This is expected to cause an earlier reactor trip on the SG low-low level. The downward adjustment of the initial SG level introduces competing effects with respect to predicted peak primary and secondary pressures and DNBR.

This event is currently not a limiting transient in this category and is bounded by the turbine trip event. Therefore, its analysis is not required. However, DPC stated that analysis may become necessary in the future due to hardware or methodology changes. In that event DPC should be required to perform sensitivity studies with respect to initial condition selections to ensure conservatism in the analysis.

3.3.2.4 Feedwater System Pipe Break

This transient is significantly impacted by implementation of the feedring steam generators, and requires three major assumption changes as a direct result of the design and location of the main feedwater nozzles. DPC's discussion of sources of assumption changes and impact of changes in transient results was reviewed and found to be reasonable.

The loss of offsite power coincident with reactor trip is assumed, resulting in RCP trip and delay in the startup of the diesel generators for SI. Early MSIV closure was determined to be conservative in terms of earlier faulted SG dryout. Thus, in the revised assumptions, MSIV closure occurs coincident with turbine trip, which occurs on loss of offsite power.

DPC's approach to analysis of this event is acceptable.

3.3.3 Decrease in Reactor Coolant System Flow Rate

Three transients analyzed in this category are: (1) partial loss of forced reactor coolant flow, (2) complete loss of forced reactor coolant flow, and (3) reactor coolant pump locked rotor.

Revisions to both of the complete and partial loss of forced RC flow are editorial changes and are acceptable.

3.3.3.1 RC Pump Locked Rotor

As stated in Section 3.2, DNBR for this event will be analyzed using the SCD methodology. Therefore, affected parameters are initially set to nominal values instead of assuming conservative values. DPC provided (Ref. 3) the explanation of the applicability of the SCD methodology for this transient and we find the explanation to be acceptable (see also Section 3.2).

DPC stated that cases with and without loss of offsite power coincident with the turbine trip will be analyzed.

As stated in the SER (Ref. [illegible]) for DPC-NE-3002 (Ref. 5), the assumption that 120% of design pressure is not an acceptable limit. DPC is required to use 110% of design pressure.

3.3.4 Reactivity and Power Distribution Anomalies

DPC added the possibility of reactor trip on high pressurizer pressure in addition to the high neutron flux for completeness.

3.3.5 Increased in Reactor Coolant Inventory

Inadvertent operation of ECCS during at-power operation is the only transient analyzed. Although DNB is a primary concern, since a potential for pressurizer overfill exists during this event, DPC added a new section to address that concern for PZR overfill leading to water relief through the PZR Safety Valves (PSVs). The acceptance criterion for this analysis is the minimum water relief temperature to assure PSV operability.

The SRP suggests use of full power unless a lower power can be justified. DPC assumes zero power (Ref. 3) in this analysis for conservatism. This is because if overfill occurs at lower initial power, then the water relief temperature is more likely to be less than the acceptance criterion. Therefore DPC selects the initial and boundary conditions in such a way to minimize relief temperature. We find this approach to be reasonable.

3.3.6 Decrease in Reactor Coolant Inventory

Inadvertent opening of a pressurizer safety or relief valve and steam generator tube rupture (SGTR) events are the two transients analyzed in this category. Proposed revisions to the inadvertent opening of a pressurizer safety or relief valve are editorial changes.

3.3.6.1 Steam Generator Tube Rupture

The steam generator tube rupture (SGTR) event was not part of the original review since the transient methodology documented in DPC-NE-3000 based on the use of the RETRAN computer was approved only for non-LOCA applications. This restriction regarding performance of SGTR analysis with RETRAN (Item vii of RETRAN SER (Ref. 11)) applies to applications which encounter two-phase flow in the primary loop, which does occur in many SGTR scenarios.

In the limited review documented in Reference 12, DPC received approval for an SGTR analysis of the worst offsite dose scenario using RETRAN for Catawba Nuclear Station Units 1 and 2. Justification was provided (Ref. 13) in a qualitative manner by DPC on each of the items cited under restrictions and limitations on the use of RETRAN in its SER. There is assurance that the use of code for that particular scenario was acceptable since DPC stated that two-phase flow was not encountered in the primary loop.

Although NRC approval was specific to Catawba units, as considered in DPC-NE-3000, Catawba and McGuire plants for the purpose of analysis qualification are interchangeable. Therefore DPC stated (Ref. 3) that NRC approval of the SGTR analysis using RETRAN should be applicable to McGuire plant analysis. We concur with DPC's statement, so long as the scenario is essentially the same and no two-phase flow conditions are encountered in the RCS primary loops.

The DNBR will be computed using the SCD methodology (see Section 3.2).

4.0 CONCLUSIONS

Revision 1 to the DPC topical report DPC-NE-3002 and the DPC responses to NRC questions and other supporting documents cited in Section 5.0 were reviewed. Review of these documents focused upon evaluation of acceptability of the proposed changes and the perceived impact of these changes.

As stated earlier, steam line break, rod ejection, dropped rod and boron dilution events were not part of this review.

Subject to the foregoing, DPC's proposed revision to approach to FSAR Chapter 15 transient analysis, as documented in Revision 1 of DPC-NE-3002 and its supporting document, was found to be acceptable subject to the following conditions:

1. The acceptability of the use of DPC's approach to FSAR analysis is subject to the conditions of SERs on all aspects of transient analysis and methodologies (DPC-NE-3000, DPC-NE-3001, DPC-NE-3002, DPC-NE-2004, DPC-NE-2005) as well the SERs on RETRAN and VIPRE computer codes.

2. There are scenarios in which an SGTR event may result in loss of subcooling and the consequent two-phase flow conditions in the primary system. In such instances, the use of RETRAN is not acceptable without a detailed review of the analysis.

3. In the future if hardware or methodology changes, selection of limiting transients needs to be reconsidered, and DPC is required to perform sensitivity studies to identify the initial conditions in such a way to avoid conflict between transient objective, such as DNB and worst primary pressure.

4. It is emphasized that, when using the SCD methodology to determine DNBR, the range of applicability of the selected CHF correlation must not be violated.

5. DPC's assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor is not acceptable: DPC is required to use 110% of design pressure for that limit.

5.0 REFERENCES

1. Letter from M.S. Tucker (DPC) to USNRC, "FSAR Chapter 15 System Transient Analysis Methodology, DPC-NE-3002," Revision 1, June 1994.

2. "Duke Power Company - The Thermal-Hydraulic Transient Analysis Methodology - Oconee Nuclear Station, McGuire Nuclear Station, Catawba Nuclear Station," DPC-NE-3000, Revision 3, August 1994.

3. Letter from M.S. Tuckman (DPC) to USNRC, "Topical Report DPC-3002, "FSAR Chapter 15 System Transient Analysis Methodology", Responses to NRC Questions," August 18, 1995.

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

5. "FSAR Chapter 15 System Transient Analysis Methodology," DPC-NE-3002, August 1991.

6. "Safety Evaluation on Topical Report DPC-NE-3000, Thermal-Hydraulic Transient Analysis Methodology," November 15, 1991.

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