GE Nuclear Energy

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NEDO-32906 Supplement 1 -A
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Licensing Topical Report

TRACG Application for Anticipated Transient Without Scram Overpressure Transient Analyses

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Approved: J.F. Klapproth
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INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-32906P, Supplement 1 -A, which has the proprietary information removed. Portions of the document that have been removed are indicated by "bars" drawn in the right-hand margin of the text (as shown to the right of this paragraph).

ACCEPTANCE VERSION ADDITIONS AND CHANGES

The NRC Safety Evaluation for this Licensing Topical Report (LTR) includes expectations regarding the content of applications that reference this LTR. Section 4.0 of the Safety Evaluation is reproduced in a new section called Application Requirements (following the Abstract) to emphasize the NRC's conditions, limitations, and expectations regarding the use of the approved LTR.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

MFN : 03-085

August 18, 2003

RECEIVED
AUG 2 1 2003
GENERAL ELECTRIC COMPANY

Mr. James F. Klapproth, Manager
Engineering & Technology
GE Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

SUBJECT: REVIEW OF GE NUCLEAR ENERGY LICENSING TOPICAL REPORT, NEDE-32906P, SUPPLEMENT 1, "TRACG APPLICATION FOR ANTICIPATED TRANSIENT WITHOUT SCRAM TRANSIENT ANALYSES" (TAC NO. MB6359)

Dear Mr. Klapproth:

By letter dated September 18, 2002, and its supplement dated July 29, 2003, GE Nuclear Energy (GENE) requested that the NRC staff review and approve its licensing topical report (LTR) NEDE-32906P, Supplement 1, "TRACG Application for Anticipated Transient Without Scram Transient Analyses." The LTR establishes an agreed-upon process and scope for the application GENE reactor accident and transient analysis computer code TRACG02A (referred to hereafter as TRACG) for the analysis of anticipated transients without scram in the operating fleet of BWR/2-6. The TRACG code is a thermal/hydraulic analysis code intended to be used in a realistic analysis mode. The approach taken by GENE in qualification of the code for the proposed application is under the Code Scaling, Applicability, and Uncertainty evaluation methodology described in NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident." The staff finds the proposed approach specified in NEDE-32906P, as supplemented, is acceptable for referencing in licensing applications to the extent specified under the limitations delineated in the report and in the associated NRC safety evaluation. The enclosed safety evaluation defines the basis for acceptance of the LTR.

The NRC requests that GENE publish an accepted version of the revised LTR within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract, and add a "-A" (designating accepted) following the report identification number (i.e., NEDE-32906P-A).

If the NRC's criteria or regulations change so that its conclusion in this letter that the LTR is acceptable is invalidated, GENE and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the LTR without revision of the respective documentation.

Pursuant to 10 CFR 2.790, we have determined that the enclosed safety evaluation does not contain proprietary information. However, we will delay placing the safety evaluation in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosed is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

If you have any questions, please contact Alan Wang, GENE Project Manager, at (301) 415-1445.

Sincerely,

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 710

Enclosure: Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NEDE-32906P, SUPPLEMENT 1, "TRACG APPLICATION FOR ANTICIPATED TRANSIENT WITHOUT SCRAM TRANSIENT ANALYSES"

GE NUCLEAR ENERGY

PROJECT NO. 710

1.0 INTRODUCTION

The staff has reviewed the GE Nuclear Energy (GENE) reactor accident and transient analysis computer code TRACG02A for application to the analysis of anticipated transients without scram (ATWS) in the operating fleet of boiling water reactors (BWR)/2-6. GENE submitted TRACG02A (referred to hereafter as TRACG) for NRC review for application to anticipated transient without scram analyses on September 18, 2002 (Reference 1), as supplemented on July 29, 2003 (Reference 2). The submittal includes the licensing topical report (LTR) on the subject. GENE previously submitted both code model documents that describe the TRACG code and the code itself to assist the staff review of the TRACG application to anticipated operational occurrences (Reference 3). The staff review and approval of that application of the TRACG code is documented in Reference 4.

The TRACG code is a thermal/hydraulic analysis code intended to be used in a realistic analysis mode. The approach taken by GENE in qualification of the code for the proposed application is under the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology described in Reference 5.

The TRAC family of computer codes began as a pressurized water reactor analysis tool developed for the NRC at the Los Alamos National Laboratory. A BWR version of the code was developed jointly by the NRC and GE at the Idaho National Engineering Laboratory as TRAC-BD-1/MOD1. GE, and later GENE, developed a proprietary version of the code designated as TRACG. The objective of the proprietary code development was to have the code capable of realistic analysis of transient, stability, and ATWS events. The code was modified to include a three-dimensional kinetics capability in addition to the multi-dimensional, two-fluid thermal/hydraulics modeling.

The plant types for which the TRACG code is to be applied includes the operating BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 designs. The code has not been submitted for review for application to any other operating plant design. The code is under separate review for application to the advanced plant design European Simplified BWR (ESBWR). This safety evaluation is applicable only to the operating BWR/2-6.

2.0 REGULATORY BASIS

The Code of Federal Regulations (10 CFR), Part 50.62, presents the requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants. The rule defines an ATWS to mean

> *...an anticipated operational occurrence as defined in appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of appendix A of this part.*

Furthermore, the rule states that, in the case of the BWR, there must be an alternate rod injection system, a standby liquid control system with the capability of injecting into the reactor pressure vessel a borated water solution, and equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS.

At this point the staff has not prepared an appropriate Standard Review Plan (Reference 6) section to provide guidance in the review of the satisfaction of the requirements detailed in 10 CFR 50.62.

The staff has prepared suggested means by which the general requirements for a thermal/hydraulics analysis computer code can be met. References 7 and 8 describe acceptable approaches by which the calculated uncertainty in the analysis methodology can be assessed. References 7 and 8 express a preference for the CSAU methodology, as the means by which uncertainty in a code calculation is to be determined.

3.0 TECHNICAL EVALUATION

The GENE TRACG code is a proprietary methodological development based on the TRAC-BD1 code developed jointly by the NRC and General Electric at the Idaho National Engineering Laboratory. The code has models and correlations in place which were developed at the commercial expense of GENE and are, thus, considered to be proprietary. The staff reviewed and approved the TRACG code for application to anticipated operational occurrences (AOOs) in the current operating fleet of BWR/2-6s. The AOOs include the increase and decrease in heat removal by the secondary system, decrease in reactor coolant flow rate, reactivity and power distribution anomalies, and increase and decrease in reactor coolant inventory, all with reactor scram. The present review of the TRACG code for application to ATWS focused on those aspects of the code which extend the prior AOO review.

The requirements of a realistic methodology are somewhat different from a prescriptive methodology in that more realistic models can be used and a measure of the uncertainty in the code must be determined. Various means of achieving an estimate of uncertainty are available in the realm of statistical analysis. GENE has chosen to follow the basic CSAU approach outlined in NUREG/CR-5249. While the CSAU approach defines the process by which uncertainty analysis is performed, it leaves room for the applicant to determine the exact statistical methodology to be applied. In both the AOO application of TRACG and the ATWS application, GENE has chosen to apply an analysis of variance statistical methodology. An

explanation of the various statistical approaches will be found in the discussion of Step 14 of the CSAU process which follows.

Comparison with the CSAU Methodology

1. Step 1. Scenario Selection

The processes and phenomena that can occur during an accident or transient vary considerably depending upon the specific event being analyzed. GENE has identified the main steam isolation valve closure (MSIVC) ATWS and pressure regulator failed open (PRFO) ATWS as the events to which the methodology under review will be applied based on previous licensing basis analysis. The MSIVC and PRFO have been found to be the limiting pressurization events in the historical analyses. Furthermore, the application of the methodology is limited to the determination of the peak vessel pressure up to the time of initiation of boron injection. The code has not been reviewed for, nor is it being approved for, evaluation of ATWS instability.

GENE is consistent with this step in the CSAU approach.

2. Step 2. Nuclear Power Plant Selection

The dominant phenomena and timing for an event can vary significantly from one nuclear power plant design to another. GENE has specified the nuclear power plant applicability for the methodology under review to be the BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 operating reactor designs.

GENE is consistent with this step in the CSAU approach.

3. Step 3. Phenomena Identification and Ranking

The behavior of a nuclear power plant undergoing an accident or transient is not influenced in an equal manner by all phenomena that occur during the event. A determination must be made to establish those phenomena that are important for each event and various phases within an event. Development of a Phenomena Identification and Ranking Table (PIRT) establishes those phases and phenomena that are significant to the progress of the event being evaluated.

The critical safety parameter for ATWS events is the peak reactor pressure vessel pressure. The main difference between an ATWS event and an AOO is that the reactor scram is not considered. The similarity, therefore, between an ATWS scenario and the AOO scenario indicates that the same phenomena apply to ATWS overpressure as for AOO with the exception of the absence of reactor scram. The PIRT developed for the AOO events is shown in Table 3-1 of Reference 3, and used by reference. The staff accepts the PIRT developed for the AOO application of TRACG as applicable to the TRACG ATWS analysis.

GENE is consistent with this step in the CSAU approach.

4. Step 4. Frozen Code Version Selection

The version of a code, or codes, reviewed for acceptance must be "frozen" to ensure that after an evaluation has been completed, changes to the code do not impact the conclusions and that changes occur in an auditable and traceable manner. GENE has specified that the TRACG02A code, which is under configuration control, was used for the ATWS analysis.

GENE is consistent with this step in the CSAU approach.

5. Step 5. Provision of Complete Code Documentation

This step is to provide documentation on the frozen code version such that evaluation of the code's applicability to postulated transient or accident scenarios for a specific plant design can be performed through a traceable record. GENE has provided the necessary documentation through reference to code documentation in the possession of the staff from the previous review of the TRACG code reported in Reference 4.

GENE is consistent with this step in the CSAU approach.

6. Step 6. Determination of Code Applicability

The applicability of the GENE methodology is addressed in the following evaluation of the technical content of the documentation.

GENE has stated in Reference 9 that the application of TRACG would be to the ATWS event up to the initiation of boron injection. The specific purpose of the code is to calculate, in a realistic fashion, the peak vessel pressure. Table 1, which was provided by GENE, identifies the event sequence for an MSIVC ATWS and compares the capabilities of the currently approved GENE methodology, ODYN, with the proposed TRACG methodology. It should be noted that the peak vessel pressure typically occurs in the first ten (10) seconds, while the startup of the standby liquid control system (SLCS) pumps to begin boron injection does not occur until approximately 120 seconds. Also, a number of additional parameters are predicted by TRACG. Since the proposed application of the methodology is restricted to prediction of the peak vessel pressure, the staff review has focused on only this aspect of the code's total capability.

The documentation (Reference 9), refers to applicability of the TRACG ATWS methodology to the operating fleet of BWR/2-6, and the MELLLA+ (NEDC-33006P) operating limits. However, MELLLA+ is not applicable to the BWR/2 class of plants. GENE clarified this point in its response to a staff's request for additional information dated July 29, 2003. This point needs to be made in the approved version of the LTR. At any rate, since MELLLA+ is still under staff review, the review of TRACG for application to the ATWS vessel peak pressure calculation has focused on the power level and reactor coolant flow rather than any specific operating limit curve. This review does not address or imply approval of MELLLA+.

Table 1
ATWS MSIVC Event Sequence
ODYN and TRACG Predictions

Key Output?	Response	Event Time (sec)	ODYN	TRACG
No	Main Steam Isolation Valve (MSIV) Isolation Initiates	0.0	Start Transient	Start Transient
No	High Pressure ATWS Setpoint	≅4	Trip Predicted	Trip Predicted
No	MSIVs Fully Closed	≅4	Modeled	Modeled
No	Peak Neutron Flux	≅4	Peak Predicted	Peak Predicted
No	Opening of First Relief Valve Tripped	≅4	Trip Predicted	Trip Predicted
No	Suppression Pool Heatup Calc Initiated	≅4	Modeled	Not Modeled
No	Recirculation Pump Tripped	≅5	Trip Predicted	Trip Predicted
Yes	Peak Vessel Pressure	≅10	Peak Predicted	Peak Predicted*
Yes	Peak Clad Temperature	≅45	Peak Predicted	Predicted but Not Used
No	Feedwater Reduction Initiated	≅20	Modeled	Modeled
Yes	Boron Injection Temperature Reached	≅40	Predicted	Not Predicted
Yes	Pre SLCS Pump Start Reactor Pressure	≅124	Predicted	Predicted
No	SLCS Pumps Start	≅124	Modeled	Transient Terminated
No	Water Level Increased	≅1700	Modeled	N/A
Yes	Hot Shutdown Achieved	≅1800	Predicted	N/A
Yes	Peak Suppression Pool Temperature	≅4000	Peak Predicted	N/A

*The transient may be terminated after peak pressure is predicted. Data after this point may be used to determine reactor pressure up until the point that SLCS injection begins.

Furthermore, while the minimum MELLLA+ flow rate has been quoted in the MELLLA+ review as 80 percent, in Reference 9 the minimum flow rate is quoted as 73 percent, a more conservative value since less energy is transferred from the fuel. While this is more conservative, individual applications of the TRACG ATWS methodology are expected to clearly state the power level used for the analysis. Should the MELLLA+ operating limit approach be approved, separately from this review, simply referring to MELLLA+ in application of the TRACG methodology to ATWS is not sufficient. The specific flow rate and power level used in the individual applications of the methodology must be clearly stated rather than reference to MELLLA+.

Reference 9 also quotes the reactor power level as 100 percent. The staff understands that to be referring to 100 percent of the plant's licensed power level, which in the case of the demonstration plant used for the application, corresponds to 113.4 percent of the original rated power. For each application of the TRACG ATWS methodology, it must be made clear exactly what power level is being used, not only the percentage of licensed power but the actual power level.

In its review of the applicability of TRACG to AOO events, the staff performed numerous comparisons between the neutronic modeling used by GENE and methodologies available to the staff (Reference 4). The staff noted that GENE uses finite difference methods while the staff uses the more modern nodal methods, which more accurately account for inter-assembly gradients. For that evaluation, three test cases were run, a slow pressure increase, an inlet flow decrease, and an MSIVC transient, all for a two fuel type core. The core specification was developed by the staff and GENE to minimize the effects of different methods used to generate the cross sections while focusing on the differences in kinetics modeling.

The first two transients were run to evaluate TRACG's ability to predict changes in total reactivity. The two methodologies compared well with each other with regard to the change in reactivity from the imposed transient. The third transient, the MSIVC, was intended to examine the GENE methodology's ability to predict prompt critical power changes. The conclusion reached based on those studies was that the GENE methods were reasonable for non-reactivity insertion accidents (RIAs). Comparison of the TRACG predicted power versus the TRAC-B/NESTLE predicted power for the MSIVC simulated pressurization transient found the TRAC-B/NESTLE peak power to be slightly more conservative than TRACG (Figure 5 in Reference 9). However, the integrated energy deposited matched very well because while the peak for TRACG was slightly lower, the drop off was slower and the tail of the power/time curve was higher. This would indicate acceptable performance for application of TRACG to the ATWS events under consideration.

Based on the results of the above noted transient analyses, the staff concluded that the TRACG kinetics code can adequately model the types of anticipated transients examined, including the MSIVC event.

Finally, application of TRACG to ATWS events assumes there is no thermal/hydraulic - neutronic instability. The methodology has not been reviewed for applicability to instability conditions.

GENE is consistent with this step in the CSAU approach.

7. Step 7. Establishment of Assessment Matrix

Support of the application of TRACG to AOOs (Reference 3), provided the assessment of the code against appropriate separate effects tests, component tests, integral systems tests, plant tests, and plant operating data. The assessment matrix adequately covered the range of conditions expected in the current application of the methodology.

GENE is consistent with this step in the CSAU approach.

8. Step 8. Nuclear Power Plant (NPP) Nodalization Definition

Reference 5 discusses the tradeoffs in determining an adequate NPP nodalization. GENE has developed guidelines for use of TRACG for transient and accident analyses that are explicit and reduce nodalization as a contributor to calculational uncertainty. The guidelines provide rules for deriving the appropriate nodalization, thus defining a method for automating the generation of input for a TRACG analysis that maintains consistency in approach from one analysis to another.

GENE is consistent with this step in the CSAU approach.

9. Step 9. Definition of Code and Experimental Accuracy

Simulation of experiments developed from Step 7 using the NPP nodalization from Step 8 provides checks to determine code accuracy. The differences between the code calculated results and the test data provide bias and deviation information. Code scale-up capability can also be evaluated from separate effects data, full scale component tests data, plant test data, and plant operating data where available. Overall code capabilities are assessed from integral systems test data and plant operational data. These assessments were performed as part of the AOO qualification of the TRACG methodology documented in Reference 3.

GENE is consistent with this step in the CSAU approach.

10. Step 10. Determination of Effect of Scale

Assessment of the TRACG AOO methodology was performed using results of separate effects tests, with both scaled and full-size components, integral facility tests, and tests performed on operating nuclear power reactors. GENE has shown in Reference 3 that full-scale plant data are bounded in the application of TRACG to AOO events. Evaluation of code results from these assessments demonstrates there is not a significant impact of scale on TRACG for the range of plants to which it is being applied.

GENE is consistent with this step in the CSAU approach.

11. Step 11. Determination of the Effect of Reactor Input Parameters and State

The purpose of this step is to determine the effect that variations in the plant operating parameters have on the uncertainty analysis. Plant process parameters characterize the state of operation and are controllable by the plant operators to a certain degree. GENE has

reviewed the operating parameters important to the progress of the ATWS event in its operating fleet of plants. The parameters identified come from the PIRT, technical specifications, and operational input.

The analyses performed were at the operating limits currently allowed and at those being reviewed by the staff for possible future operation. The staff notes that the flow rate/power condition proposed as MELLLA+ is still under staff review. The approval granted herein for use of TRACG for calculation of the reactor pressure vessel peak pressure under ATWS conditions is in no way an approval of the use of MELLLA+. The review and approval are consistent with a power condition and a flow rate. In addition, the MELLLA+ operating limits line is not applicable to the BWR/2 line of plants.

GENE is consistent with this step in the CSAU approach, but approval of the use of TRACG for ATWS reactor pressure vessel peak pressure does not imply approval of the MELLLA+ operating limits.

12. Step 12. Performance of NPP Sensitivity Calculations

Sensitivity calculations are typically performed to evaluate methodology sensitivity to parameters predicted by the methodology, and to various plant operating conditions that arise from uncertainties in the reactor state at the initiation of the transient. In the case of ATWS analyses, due to the extremely low probability of an ATWS event, the staff has accepted nominal plant parameters for ATWS analysis. For ATWS analysis, technical specification allowable values have been used for analytic limits. GENE procedures require that both GENE and the licensee agree to the limiting input values. When individual applications of the methodology are submitted for staff review, the critical ATWS plant parameters will be reviewed for consistency with plant design and technical specification allowable limits.

GENE is consistent with this step in the CSAU approach.

13. Step 13. Determination of Combined Bias and Uncertainty

The individual uncertainties resulting from code models of important phenomena, scale effects, and NPP input parameter variations must be combined to obtain an overall bias and uncertainty.

GENE provided sample calculations of the MISVC and PRFO events. The models used are taken as demonstrations of the applicability of the TRACG methodology to the ATWS event in the operating fleet of BWRs.

GENE is consistent with this step in the CSAU approach.

14. Step 14. Determination of Total Uncertainty

The first few steps in the CSAU methodology identify and rank the physical phenomena important to judging the performance of the safety systems and margins in the design. The phenomena are compared to the modeling capability of the code to assess whether the code has the necessary models to simulate the phenomena. Most important, the range of the

identified phenomena covered in experiments or test data is compared to the corresponding range of the intended application to assure that the code has been qualified for the highly ranked phenomena over the appropriate range. The result is then provided in a PIRT. The staff has reviewed the PIRT provided for TRACG in References 9 and 3 by inclusion, and finds it acceptable and consistent with the staff's experience in judging the important phenomena associated with the ATWS event in BWRs.

The discussion of the uncertainty analysis approach presented in References 5 and 10 envisioned the use of response surfaces for quantifying uncertainty. The staff recognizes that there are other valid and acceptable means by which the uncertainty can be assessed. Other means include that developed following the work by Wilks (Reference 11), referred to as order statistics, and a related method referred to as analysis of variance. Each has advantages and disadvantages.

Briefly, the statistical methods can be described as follows.

Response Surface:	Response surface for the safety parameter is generated from parameter perturbations of each parameter singly. Statistical upper bound is determined from the Monte Carlo method using a response surface.
Order Statistics:	Monte Carlo method using random perturbations of all important parameters is done at once. Sample size defined to yield desired statistical confidence. Statistical upper bound is determined from most limiting perturbation (for the first order statistics).
Normal Distribution:	Monte Carlo method using random perturbations of all important parameters is done at once. Normality of output distribution established by statistical checks. Statistical upper bound is determined from sample variance from all perturbations.

The GENE methodology as described in Reference 8, applies a statistical method based on normal distribution-one-sided upper tolerance limit (ND-OSUTL). As noted above, the staff recognizes that there are various means by which uncertainty in a code calculation can be obtained. While the staff has referred to the response surface approach in References 5 and 10, there are advantages in use of other methods such as ND-OSUTL. Most notably, the methodology can require significantly reduced calculational cases, thus reducing the cost of performing an analysis. But, there are significant restrictions, such as demonstration of normality of the resulting probability distribution function. There is not one single correct method of determining uncertainty. Each of these methods has been reviewed carefully by the staff in various submittals that have been reviewed in recent years.

The acceptability of the ND-OSUTL approach is discussed further in Reference 4.

GENE has determined the mean vessel peak pressure by using Monte Carlo sampling of the applicable internal model parameters and initial condition biases and uncertainties to produce an overall peak pressure bias and uncertainty. By using 39 trials (giving a probability of 95 percent at a confidence level of 87 percent, from $\beta = 1 - \gamma^N$, where β is the confidence level, γ is

the probability and N is the number of samples) a peak pressure distribution has been obtained which has been shown to be normal by application of the Anderson-Darling test. For the test plant, the MSIVC event returns a vessel peak pressure of 1415 pounds per square inch (psi) and the PRFO a peak vessel pressure of 1391 psi.

GENE is consistent with this step in the CSAU approach.

4.0 CONDITIONS AND LIMITATIONS

Based on review of the proposed application of the TRACG methodology to analysis of the ATWS event in the current GENE operating fleet, that is BWR/2-6 plants, the following conditions on use of the methodology are imposed.

1. Application of the methodology is considered for prediction of the reactor vessel peak pressure only. The prediction is to be terminated at the time of the signal to initiate SLCS pump injection of boron into the reactor coolant system.

2. Simply referring to MELLLA+ in the application of the TRACG methodology to ATWS is not sufficient. The flow rate and power level used in the individual applications must be clearly stated and the power-to-flow ratios must not be outside the ranges used in this review. MELLLA+ is not applicable to the BWR/2 class of plants. This point needs to be made in the approved version of the LTR.

3. For each application of the TRACG ATWS methodology, it must be made clear exactly what power level is being used, not only the percentage of licensed power, but the actual power level.

4. Application of TRACG to ATWS events assumes there is no thermal/hydraulic - neutronic instability. The methodology has not been reviewed for applicability to instability conditions.

The staff also notes that a generic topical report describing a code such as TRACG cannot provide full justification for each specific individual plant application. When a licensee proposes to reference the TRACG-based ATWS methodology for use in a license amendment, the individual licensee or applicant must provide justification for the specific application of the code in its request which is expected to include:

1. Nodalization: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

2. Chosen Parameters and Conservative Nature of Input Parameters: A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.

3. Calculated Results: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure.

5.0 CONCLUSIONS

The staff concludes from its review of the documentation and code submitted that the TRACG methodology when applied to the prediction of the ATWS vessel peak pressure is structured consistent with the CSAU methodological process, and satisfactorily reflects the intended use of the methodology to address licensing requirements for the event in the operating fleet of BWR/2s, BWR/3s, BWR/4s, BWR/5s and BWR/6.

The review resulted in conditions on use of the code already noted in this safety evaluation. In particular, since MELLLA+ is still under staff review, the review of TRACG for application to the ATWS vessel peak pressure calculation has focused on the power level and reactor coolant flow rather than any specific operating limit curve. This review does not address or imply approval of MELLLA+.

6.0 REFERENCES

1. Letter from George Stramback (GENE) to U.S. NRC dated September 18, 2002.

2. Letter from George Stramback (GENE) to U.S. NRC, "Request for Additional Information Related to the Review of GE Licensing Topical Report NEDE-32906P Supplement 1, 'TRACG Application for Anticipated Transient Without Scram Transient Analyses'," dated July 29, 2003.

3. NEDE-32906P, Rev. 0, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," January 2000.

4. Safety Evaluation Report by the Office of Nuclear Reactor Regulation for NEDE-32906P "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," June 2002.

5. NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," December 1989.

6. NUREG-0800, "United States Nuclear Regulatory Commission Standard Review Plan," Revisions as of January 1, 2002.

7. Draft Regulatory Guide, DG-1120, "Transient and Accident Analysis Methods," U.S. NRC, December 2000.

8. Draft Standard Review Plan, Section 15.0.2, "Review of Analytical Computer Codes," U.S. NRC, December 2000.

9. NEDE-32906P Supplement 1, "TRACG Application for Anticipated Transient Without Scram Transient Analyses," September 2002.

10. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1999.

11. Wilks, S. S., "Determination of Sample Sizes for Setting Tolerance Limits," Ann. Math. Statistics, Vol. 12, 1941.

Principal Contributor: R. Landry

Date: August 18, 2003

ACKNOWLEDGMENTS

The TRACG ATWS Overpressure Application is the result of the technical contributions from many individuals. J. S. Stewart, C. L. Heck, L. A. Klebanov, R. N. Anderson, and E. W. Gibbs have made significant contributions to the development of the TRACG ATWS Overpressure Application. The advice and careful review from B. S. Shiralkar, J. L. Casillas, R. H. Jacobs, Y. C. Chu, W. Marquino, J. E. Leatherman, and J. G. M. Andersen is highly appreciated.

ACRONYMS AND ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
AL	Analytical Limit
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
BWR/*n*	GE BWR product line *n* (*n* can be 2, 3, 4, 5, or 6)
CSAU	Code Scaling, Applicability, and Uncertainty
CTR	Customer Technical Requirement
EOC	End of Cycle
GE	General Electric Company
GE13	GE fuel product line 13
GE14	GE fuel product line 14
GESTAR	GE Standard Application for Reactor Fuel
K	Kelvin
kPa	kilo-Pascal
LOCA	Loss of Coolant Accident
LTR	Licensing Topical Report
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MOC	Middle of Cycle
MPa	Mega-Pascal
MSIV	Main Steam Isolation Valve
MSIVC	MSIV Closure ATWS Event
MSIVF	MSIV Closure with Flux Scram AOO
MWth	Mega-Watt Thermal
NRC	United States Nuclear Regulatory Commission
ODYN	One-Dimensional Reactor Dynamics Code
OSUTL	One Sided Upper Tolerance Limit
$OSUTL_{x/y}$	One Side Upper Tolerance Limit with x% content at y% confidence

PANACEA	Three-Dimensional BWR Core Steady State Simulator Code
PCT	Peak Clad Temperature
PRFO	Pressure Regulator Failed Open ATWS Event
psi	pounds per square inch
psig	pounds per square inch - gauge
RHR	Residual Heat Removal
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
S/RV	Safety/Relief Valve
SBWR	Simplified Boiling Water Reactor
SER	Safety Evaluation Report
SRP	Standard Review Plan
TASC	Single Channel Transient Analysis Code
TRACG	GE version of the Transient Reactor Analysis Code

CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABSTRACT

This report discusses the application of TRACG, the General Electric (GE) proprietary version of the Transient Reactor Analysis Code, to analyses of Anticipated Transient Without Scram (ATWS) peak vessel pressure for Boiling Water Reactors (BWRs). Realistic calculations with TRACG can be used to support licensing evaluations for these transient events. The information presented in this report is an extension to the information reviewed and approved for Anticipated Operational Occurrences (AOOs).

APPLICATION REQUIREMENTS

The NRC Safety Evaluation (SE) for this Licensing Topical Report (LTR) includes expectations regarding the content of applications that reference this LTR. Section 4.0 of the SE is reproduced here to emphasize the conditions, limitations, and expectations regarding the use of the approved TRACG Application for Anticipated Transient Without Scram Overpressure Transient Analyses.

> Based on review of the proposed application of the TRACG methodology to analysis of the ATWS event in the current GENE operating fleet, that is BWR/2-6 plants, the following conditions on use of the methodology are imposed.
>
> 1. Application of the methodology is considered for prediction of the reactor vessel peak pressure only. The prediction is to be terminated at the time of the signal to initiate SLCS pump injection of boron into the reactor coolant system.
> 2. Simply referring to MELLLA+ in the application of the TRACG methodology to ATWS is not sufficient. The flow rate and power level used in the individual applications must be clearly stated and the power-to-flow ratios must not be outside the ranges used in this review. MELLLA+ is not applicable to the BWR/2 class of plants. This point needs to be made in the approved version of the LTR.
> 3. For each application of the TRACG ATWS methodology, it must be made clear exactly what power level is being used, not only the percentage of licensed power, but the actual power level.
> 4. Application of TRACG to ATWS events assumes there is no thermal/hydraulic - neutronic instability. The methodology has not been reviewed for applicability to instability conditions.
>
> The staff also notes that a generic topical report describing a code such as TRACG cannot provide full justification for each specific individual plant application. When a licensee proposes to reference the TRACG-based ATWS methodology for use in a license amendment, the individual licensee or applicant must provide justification for the specific application of the code in its request which is expected to include:
>
> 1. Nodalization: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.
> 2. Chosen Parameters and Conservative Nature of Input Parameters: A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.
> 3. Calculated Results: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure.

1.0 INTRODUCTION

1.1 Background

Reference [1] provides the licensing basis for the TRACG application to Anticipated Operational Occurrences (AOOs). The U. S. Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) for Reference [1] applies the following condition and limitation on the use of the TRACG code for analysis of AOO events: "TRACG is not acceptable for application to ATWS analyses without specific staff review".

The ATWS analyses are performed with ODYN in accordance with Reference [5]. The ODYN code along with the TASC code [9] is used to determine peak vessel pressure and Peak Clad Temperature (PCT) [3]. For the suppression pool heat-up, the method includes an energy balance on the suppression pool, considering the Safety/Relief Valve (S/RV) steam flow, Residual Heat Removal (RHR) heat exchanger capacity, initial suppression pool conditions, and service water temperature.

1.2 Summary

This document demonstrates the acceptable use of TRACG analysis results for licensing BWR/2-6 power plants within the applicable licensing bases. GE has provided information to support *the use of TRACG as an alternative to previously approved methods of analyzing BWR ATWS with respect to the peak vessel pressure acceptance criterion.* This application specifically addresses TRACG capabilities to provide reasonable assurance that applicable reactor coolant pressure boundary limits are not exceeded during an ATWS.

1.3 Scope of Review

2.0 LICENSING REQUIREMENTS AND SCOPE OF APPLICATION

2.1 10CFR50 Appendix A

Anticipated Transient Without Scram (ATWS) means an Anticipated Operational Occurrence (AOO) as defined in this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of 10CFR50.

2.2 Standard Review Plan Guidelines (NUREG 800)

ATWS events and acceptance criteria are not identified in the Standard Review Plan [2].

2.3 Current Implementations and Practices

NRC Staff Report for ATWS events is described in NUREG–0460 [10].

2.4 Proposed Application Methodology

2.4.1 Conformance with CSAU Methodology

The proposed application methodology using TRACG for ATWS Overpressure transient analyses addresses all the elements of the NRC-developed CSAU evaluation methodology [6]. The CSAU report describes a rigorous process for evaluating the total model and plant parameter uncertainty for a nuclear power plant calculation. The rigorous process for applying realistic codes and quantifying

the overall model and plant parameter uncertainties appears to represent the best available practice. While the CSAU methodology was developed for application to Loss-Of-Coolant Accidents (LOCAs), there are no technical reasons that prevent CSAU methodology from being applied to other event scenarios such as ATWS overpressure.

2.4.2 Advantages of TRACG Compared to the Current Process

The primary advantage of TRACG over the current process used for ATWS peak pressure transient analyses is:

2.5 Implementation Requirements

The implementation of TRACG into actual licensing analysis is contingent on completion of the following implementation requirements:

- Review and approval by the NRC of the process for analyzing ATWS overpressure events described in Section 7.0.

2.6 Review Requirements For Updates

The review requirements for updates are described in Reference [1], Section 2.6.

2.7 ATWS Scenario Specification

The transient scenarios are those associated with ATWS in BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 type plants are the same as assumed for AOO transient events. Specifically:

1. Pressurization events, including: turbine trip without bypass, load rejection without bypass, feedwater controller failure increasing flow, pressure regulator failed closed, main steam line isolation valve closure. This grouping includes all events in SRP Section 15.2.1 - 15.2.5 which apply to BWRs. The feedwater controller failure increasing flow is in Section 15.1.1 - 15.1.4 but can also be considered a pressurization transient. The loss of auxiliary power is in SRP Section 15.2.6.
2. Depressurization events, including: pressure regulator failed open, relief valve opening. The pressure regulator failed open is in SRP Section 15.1.1 - 15.1.4. The inadvertent relief valve opening is in SRP Section 15.6.1.

3. Core flow transients, including: pump trips, startup of idle pumps, pump runup or rundown, flow control valve actuations. This grouping includes all events in SRP Section 15.3.1 - 15.3.5 and 15.4.4 - 15.4.5 which apply to BWRs.
4. Cold water events, including: loss of feedwater heating and inadvertent high pressure coolant injection. The loss of feedwater heating (decrease in feedwater temperature) is in SRP Section 15.1.1 - 15.1.4. This grouping includes all events in SRP Section 15.5.1 - 15.5.2 which apply to BWRs.
5. Level transient events such as partial or complete loss of feedwater. This grouping includes all events in SRP Section 15.2.7 that apply to BWRs.

A detailed description of the MSIVC and PRFO events is given in Section 8.0.

2.8 Nuclear Power Plant Selection

The included plant types are BWR/2s, BWR/3s, BWR/4s, BWR/5s, and BWR/6s. Both jet pump and non-jet pump designs are included. For the jet pump designs, the recirculation flow control systems include motor-generator designs, flow control valve designs, and variable speed pump designs.

3.0 PHENOMENA IDENTIFICATION AND RANKING

The critical safety parameter, for ATWS overpressure transients, is peak reactor pressure vessel (RPV) pressure. Due to the similarity between ATWS peak RPV pressure and AOO peak RPV pressure, the same phenomena apply to ATWS overpressure as for AOO (excluding scram phenomena). These parameters are shown in Table 3-1 of Reference [1].

4.0 APPLICABILITY OF TRACG TO ATWS OVERPRESSURE

Important BWR phenomena have been identified and TRACG models have been developed to address these phenomena as indicated in Table 4-1 of Reference [1].

5.0 MODEL UNCERTAINTIES AND BIASES

All high and medium ranked phenomena for pressurization and depressurization events (Table 3-1 of Reference [1]) are considered for ATWS overpressure scenarios (MSIVC and PRFO). Those parameters that do not apply to the ATWS overpressure scenarios have been excluded from the ATWS MSIVC and PRFO statistical basis (B11, C1CX, and C18).

For the high and medium ranked phenomena, the bases used to establish the nominal value, bias and uncertainty for that parameter are documented in Section 5.1 of Reference [1].

The uncertainty screening results for the Main Steam Isolation Valve Closure ATWS (MSIVC) event are shown in Section 8.3.1.1. The uncertainty screening results for the Pressure Regulator Failed Open ATWS (PRFO) event is shown in Section 8.3.1.2.

The Effect of Nodalization and Effect of Scale is described in Section 5.2 and 5.3 of Reference [1].

6.0 APPLICATION UNCERTAINTIES AND BIASES

6.1 Input

Specific inputs for each transient event are specified via internal procedures, which are the primary means used by GE to control application of engineering computer programs. The specific code input will be developed in connection with the Application Licensing Topical Report (LTR), the NRC SER and the development of the application specific procedure. This section will be limited to a more general discussion of how input is treated with respect to quantifying the impact on the calculated results. As such, it serves as a basis for the development of the application specific procedures.

Code inputs can be divided into four broad categories: (1) geometry inputs; (2) model selection inputs; (3) initial condition inputs; and (4) plant parameters. For each type of input, it is necessary to specify the value of the input. A discussion of categories (1) and (2) is contained in Section 6.1 of Reference [1]. Since initial conditions and plant parameters will be handled slightly differently for ATWS overpressure than for AOOs, Section 6.2 and Section 6.3 provide the basis for ATWS initial conditions and plant parameters.

6.2 Initial Conditions

As described in Section 6.2 of Reference [1], *initial conditions* are those conditions that define a steady-state operating condition. Initial conditions may vary due to the allowable operating range or due to uncertainty in the measurement at a given operating condition. The key plant initial conditions and associated uncertainties are given in Table 6-1 of Reference [1].

Due to the extremely low probability of the occurrence of an ATWS, the NRC Staff has accepted nominal initial conditions for ATWS analysis. However, as previously mentioned, defining a nominal initial condition is not always straightforward. Consequently, the transients will be initiated from the limiting point(s) in the allowed operating domain. Specifically, the impact of a particular initial condition on the results is characterized in the following manner:

- The results are sensitive to the initial condition and a basis for the limiting initial condition can not be established. Future plant analyses will consider the full allowable range of the initial condition.
- The results are sensitive to the initial condition and a basis for the limiting initial condition can be established. Future plant analyses will consider the parameter to be at its limiting initial condition.
- The results are not sensitive to the initial condition and a nominal initial condition will be assumed for the parameter.

6.3 Plant Parameters

A *plant parameter* is defined as a plant-specific quantity such as a protection system setpoint, valve capacity or stroke time, or a scram characteristic, etc. *Plant parameters* influence the characteristics of the transient response and have essentially no impact on steady-state operation, whereas *initial conditions* are what define a steady-state operating condition.

GE procedures for Customer Technical Requirements (CTRs) require that both GE and the Licensee agree to design input. All critical ATWS plant parameters will be reviewed in this manner.

7.0 APPLICATION METHODOLOGY

This section describes the basis for an application methodology that requires performance of a nominal calculation (*i.e.*, using nominal TRACG internal model parameters) with bounding initial conditions and plant parameters.

Section 7.1 addresses the reasonableness of performing analyses with nominal model parameters.

Section 7.2 addresses the basis for consideration of initial conditions and plant parameters.

Section 7.3 quantifies the conservatisms in the model parameters.

7.1 Determination of Mean Peak Vessel Pressure

A nominal calculation should be shown to adequately represent the mean result, given the potential distribution of model parameters (not all model parameters are centered about the mean value of the parameter) and the potential asymmetry of effect in model parameter perturbations (the magnitude of peak pressure perturbation may be different for a $-1\ \sigma$ model parameter perturbation than a $+1\ \sigma$ model parameter perturbation).

A statistical analysis is performed to demonstrate the reasonableness of the nominal approach. Specifically, as described in Section 7.0 of Reference [1], a Monte Carlo technique is used to combine the applicable individual internal model and initial condition biases and uncertainties into an overall peak pressure bias and uncertainty. Thirty nine trials are performed with the applicable parameters randomly perturbed according to their individual probability distribution functions, resulting in a distribution of code output peak vessel pressures (39 was judged to be a sufficient

number of trials to produce a reasonably accurate estimate of the population standard deviation). The results of the statistical analysis for MSIVC ATWS are presented in Figure 7-1.

Figure 7-1. MSIVC Peak Vessel Pressure Difference from Nominal (kPa) Descriptive Statistics

7.2 Consideration of Initial Conditions and Plant Parameters

The limiting initial conditions are demonstrated in the Section 8.2.1 analyses. The key sensitivities are as expected. These results are consistent with analyses performed in accordance with Reference [5]. The demonstration results can be applied to other plant types and fuel types based on these results and analysis experience.

The results of the initial conditions uncertainty (Section 8.2.2) are only important for purposes of evaluating the statistical distribution described in Section 7.1.

The key plant parameter sensitivities are demonstrated in the Section 8.2.3 analyses. The key sensitivities are as expected. These results are consistent with analyses performed in accordance with References [3] and [5]. The demonstration results can be applied to other plant types and fuel types based on these results and analysis experience.

These analyses will not be repeated for licensing calculations unless it is determined that the range and scope of the analyses presented are insufficient to characterize a particular plant or fuel type.

7.3 Quantification of Method Conservatism

These analyses demonstrate that significant conservatism is included by applying bounding initial conditions and plant parameters. Further, the effect of model uncertainties is small compared to this conservatism. This conclusion can be applied to other plant types and fuel types based on these results and analysis experience.

8.0 DEMONSTRATION ANALYSIS

The TRACG performance is demonstrated on the MSIVC and PRFO scenarios specified in Section 2.7. This demonstration includes:

1. a TRACG baseline analysis for a representative plant and fuel type,
2. a demonstration of the sensitivity of the transient to initial conditions and plant parameters,
3. a demonstration of the sensitivity of the transient to the individual model uncertainties, and
4. a demonstration of the statistical distribution of results.

8.1 Baseline Analysis

Figure 8-1. TRACG Core Map

Table 8-1. TRACG Channel Grouping

8.1.1 MSIV Closure ATWS (MSIVC) Baseline Analysis

The steam line isolation causes a rapid increase in reactor vessel pressure, which results in core void reduction. Consequently, power increases with positive void reactivity insertion. For ATWS simulation purposes, the expected MSIV position and high flux scrams do not occur. The power excursion is initially mitigated by void production from the increased core heat flux, as well as negative doppler reactivity from increasing fuel temperature. Soon after the time the MSIVs are fully closed, Recirculation Pump Trip (RPT) is initiated on high pressure, such that core flow begins to decrease. At about this same time, the Safety/Relief Valves (S/RVs) open, reducing the rate of pressure increase. As core flow continues to decrease, core voiding increases, causing the power to decrease in parallel. Finally, the steam production decreases to the point at which the S/RV capacity is sufficient to relieve all of the steam generation, and the pressure begins to fall. The key parameters are presented in Figures 8-2 through 8-4 and Table 8-2 for the MSIVC event.

Figure 8-2. TRACG Power and Flow Response for MSIVC Event

Figure 8-3. TRACG Pressure and Relief Valve Response for MSIVC Event

Figure 8-4. TRACG Vessel Inlet and Exit Flow for MSIVC Event

Table 8-2. MSIVC Key Transient Parameters

8.1.2 Pressure Regulator Failed Open ATWS (PRFO) Baseline Analysis

The pressure regulator failed open transient is initiated by setting the pressure regulator output to the upper limit at transient time 0.0. This causes the turbine control valves and turbine bypass valves to fully open, increasing steam flow, and dropping system pressure. The depressurization causes the level to swell and power to decrease due to increased core voids. The level swell may result in a turbine trip and feedwater pump trip. However, these trips, as well as all scrams, are disabled for purposes of the ATWS simulation (the pressurization caused by a turbine trip is bounded by the pressurization caused by the MSIV closure). Eventually, the pressure drops to the point at which a steam line isolation is initiated, effectively causing an MSIVC from a transient starting point. The key parameters are presented in Figures 8-5 through 8-7 and Table 8-3 for the PRFO event.

Figure 8-5. TRACG Power and Flow Response for PRFO Event

Figure 8-6. TRACG Pressure and Relief Valve Response for PRFO Event

Figure 8-7. TRACG Vessel Inlet and Exit Flow for PRFO Event

Table 8-3. PRFO Key Transient Parameters

8.2 Initial Condition and Plant Parameter Review

8.2.1 Initial Conditions and Allowable Operating Range

As described in Section 6.2 of Reference [1], the impact of the initial condition on the results are characterized in the following manner:

A. The results are sensitive to the initial condition and a basis for the limiting initial condition can not be established. Future plant analyses will consider the full allowable range of the initial condition.

B. The results are sensitive to the initial condition and a basis for the limiting initial condition can be established. Future plant analyses will consider the parameter is at its limiting initial condition.

C. The results are not sensitive to the initial condition and a nominal initial condition will be assumed for the parameter.

Table 8-4. Allowable Operating Range Characterization Basis

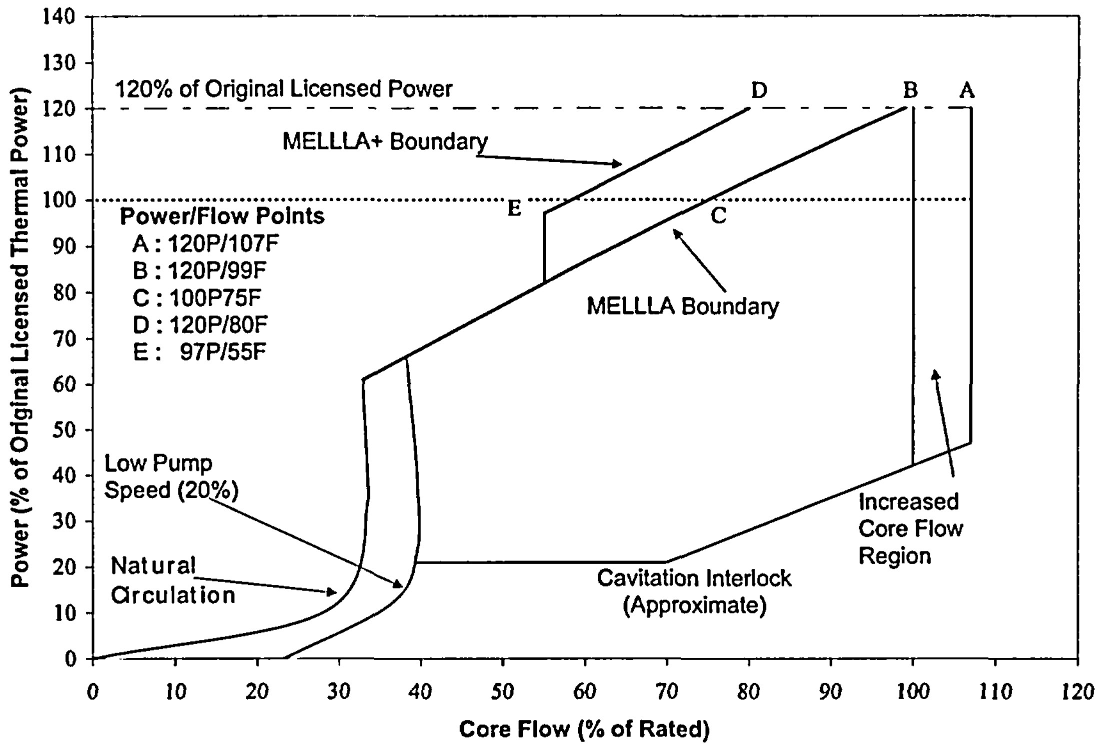

Figure 8-8. Typical Power/Flow Map

Figure 8-9. Axial Power Shape

8.2.1.1 MSIVC Allowable Operating Range Results

A summary of the sensitivity analysis for the MSIVC transient is provided in Table 8-5.

Table 8-5. MSIVC Allowable Operating Range Results

The characterization of these results is described in Table 8-6.

Table 8-6. MSIVC Allowable Operating Range Characterizations

Based on the analysis results, all trends could be characterized. Where applicable, the application procedure will require analysis at the limiting initial condition.

8.2.1.2 PRFO Allowable Operating Range Results

A summary of the sensitivity analysis for the PRFO transient is provided in Table 8-7.

Table 8-7. PRFO Allowable Operating Range Results

The characterization of these results is described in Table 8-8.

Table 8-8. PRFO Allowable Operating Range Characterizations

Based on the analysis results, all trends could be characterized. Where applicable, the application procedure will require analysis at the limiting initial condition.

8.2.2 Initial Conditions Uncertainty

As described in Section 6.2 of Reference [1], the initial condition is monitored through the use of plant sensors or on-line calculations based on plant sensors. Because of instrument or simulation uncertainty, the plant condition may vary from the indicated value. The results are characterized in the following manner:

A. The results are sensitive to the uncertainty in the initial condition. The uncertainty is included in the statistical analysis.

B. The results are not sensitive to the uncertainty in the initial condition.

With the exception of core power, the results from the allowable operating range evaluations, documented in Section 8.2.1, are used for the characterization. A summary of the results of the core power sensitivity analysis for the MSIVC transient is provided in Table 8-9. A summary of the results of the core power sensitivity analysis for the PRFO transient is provided in Table 8-10.

Table 8-9 . MSIVC Initial Condition Uncertainty Results

Table 8-10 . PRFO Initial Condition Uncertainty Results

The characterization of these results is described in Table 8-11.

Table 8-11. Initial Condition Uncertainty Characterizations

8.2.3 Plant Parameters

Table 8-12. MSIVC MSIV Closing Stroke Time Sensitivity Results

Table 8-13. PRFO Steam Line Isolation Setpoint Sensitivity Results

Table 8-14 . MSIVC S/RV Capacity Sensitivity Results

Table 8-15 . PRFO S/RV Capacity Sensitivity Results

8.2.4 Summary of Initial Conditions and Plant Parameters

The conclusions from the initial conditions and plant parameter analysis form the basis of the plant specific analysis process. The following can be concluded based on the initial condition and plant parameter analysis results:

8.3 Statistical Analysis for Licensing Events

8.3.1 Uncertainty Screening

8.3.1.1 MSIVC Uncertainty Screening

Analyses have been performed at both the +1 σ and -1 σ level for each of the model uncertainties. These results are presented in Figure 8-10.

Figure 8-10. MSIVC ATWS Peak Vessel Pressure Sensitivity to Individual Uncertainties (Pcase-Pnominal [kPa])

8.3.1.2 PRFO Uncertainty Screening

Analyses have been performed at both the +1 σ and -1 σ level for each of the model uncertainties. These results are presented in Figure 8-11.

Figure 8-11. PRFO ATWS Peak Vessel Pressure Sensitivity to Individual Uncertainties (Pcase-Pnominal [kPa])

8.3.2 Statistical Trials

A set of thirty nine statistical trials are performed for the MSIVC ATWS transient. For each trial, the model parameters listed in the uncertainty screening (Section 8.3.1.1), as well as the initial power, are randomly sampled from each parameter's prescribed probability distribution function. The descriptive statistics for the resultant distribution of peak vessel pressure difference from nominal are given in Figure 7-1.

9.0 CONCLUSIONS

- TRACG is capable of simulating ATWS overpressure events.
- The nominal TRACG calculation adequately represents the mean of the peak pressure distribution which results from considering model and initial condition uncertainties.
- The nominal TRACG calculation, combined with bounding initial conditions and plant parameters, produces an overall conservative estimate of ATWS peak vessel pressure.

10.0 REFERENCES

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