March 22, 2000

Mr. William A. Eaton
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF ALTERNATE SOURCE TERM LIMITED SCOPE APPLICATION FOR THE TIMING OF THE ONSET OF GAP ACTIVITY RELEASE (TAC NO. MA4252)

Dear Mr. Eaton:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 143 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS). This amendment authorizes revision of the GGNS Updated Final Safety Analysis Report (UFSAR) in response to your application dated November 3, 1998, supplemented by your letter dated October 7, 1999.

The amendment authorizes a limited-scope application of NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," alternative source term insights. The amendment allows a change in the minimum time assumed for the onset of fission product release from perforated fuel rods, i.e., gap activity release, following a postulated design basis loss-of-coolant accident. The timing of the gap activity release may now be delayed by up to 121 seconds instead of the instantaneous release currently specified in Section 15.6.5.5.2 of the GGNS UFSAR.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/
S. Patrick Sekerak, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 143 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Entergy Operations, Inc. (the licensee) dated November 3, 1998, as supplemented by letter dated October 7, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 143 , the license is amended to authorize revision of the Updated Final Safety Analysis Report (UFSAR) as set forth in the application for amendment by Entergy Operations, Inc. dated November 3, 1998, as supplemented by letter dated October 7, 1999. Entergy Operations, Inc. shall revise the UFSAR to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e). Implementation of the amendment is the incorporation into the UFSAR, the changes to the description of the facility as described in the licensee's application dated November 3, 1998, as supplemented by letter dated October 7, 1999, and evaluated in the staff's safety evaluation enclosed with this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gram

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the UFSAR

Date of Issuance: March 22, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Revise the Grand Gulf Nuclear Station Updated Final Safety Analysis Report, Section 15.6.5.5.2, to read as follows:

15.6.5.5.2 Containment Activity Inventory

The activity released from the severely damaged core enters the drywell at 121 seconds after the accident. This timing assumption recognizes conclusions derived from source term studies as described in NUREG-1465. Transfer from the drywell to the containment is either through the suppression pool, where a decontamination factor of 10 is taken, or through drywell leakage, which bypasses the suppression pool. This bypass flow is assumed to be equally divided between containment regions 1, 3, and 4 defined below. The flowrates for each of these drywell release pathways is based on the pressure differential between the drywell and containment (see Section 6.2). Suppression pool scrubbing (with a DF [decontamination factor] of 10) is assumed to remain effective as long as there is flow from the drywell into the suppression pool.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated November 3, 1998, as supplemented by letter dated October 7, 1999, Entergy Operations, Inc., et al. (the licensee) submitted a request for a change to the Grand Gulf Nuclear Station, Unit 1 (GGNS) licensing basis for the release of fission products following an accident. The proposed licensing basis change makes use of one of the insights established in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 1). The GGNS application credits the NUREG-1465 insight that there is a delay in the release of fission products from a perforated fuel rod, i.e., gap activity release, following a postulated design-basis accident. The requested change involves a limited-scope application of the NUREG-1456 alternate source terms by addressing only the timing of gap activity release.

The timing and duration of the gap activity release insight requested here involves a change to accident dose consequence analysis assumptions that had been made consistent with TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 2). No other assumptions or methodology changes are proposed. While the TID-14844 standard is referenced by 10 CFR Part 100, it is referenced as a guide. The dose consequences of this change have been analyzed by the licensee for conformance to 10 CFR Part 100 and 10 CFR Part 50, Appendix A, General Design Criteria 19 guidelines. No exemption from these regulations is required to support this change.

Specifically, the change would revise the GGNS Updated Final Safety Analysis Report (UFSAR), Section 15.6.5.5.2, to increase the minimum time for gap activity release to 121 seconds, instead of the assumption of instantaneous release, based on TID-14844 guidance, currently specified. The 121-second delay in gap activity release is based on analyses sponsored by the Boiling Water Reactor Owners Group (BWROG) documented in the General Electric (GE) Company Report, "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR [Boiling Water Reactor]," (Reference 3). The BWROG/GE report has been previously reviewed and approved by the Nuclear Regulatory Commission (the staff, NRC, or the Commission) (Reference 4).

2.0 BACKGROUND

In SECY-96-242, "Use of the NUREG-1465 Source Term at Operating Reactors," dated November 25, 1996, the staff informed the Commission of its approach to allow the use of the revised accident source term described in NUREG-1465 at operating plants. In the SECY paper, the staff also described its plans to (1) undertake a rebaselining assessment of two nuclear power plants to further evaluate the issues involved with applying the revised accident source term at operating plants, (2) review the pilot plant applications implementing the revised accident source terms following completion of the rebaselining effort, and (3) incorporate the total effective dose equivalent methodology in the review of the pilot plant applications. The Commission approved these plans and directed the staff to commence rulemaking upon completion of the rebaselining and concurrent with the pilot plant reviews.

The staff has completed its rebaselining effort and presented the results in SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," dated June 30, 1998. The staff submitted a rulemaking plan for implementation of the revised source term at operating reactors in SECY-98-158, "Rulemaking Plan for Implementation of Revised Source Term at Operating Reactors," dated July 30, 1998. In response to this plan, the Commission directed the staff to allow limited or selective application of the revised source term at operating reactors and promptly complete review of the pilot plant initiatives. Accordingly, the staff has initiated review of submittals from pilot plants.

GGNS is a lead pilot plant requesting selective implementation (fission product release timing only) of the revised accident source term presented in NUREG-1465. NUREG-1465 estimated the minimum time to the onset of gap activity release to be about 30 seconds unless plant-specific calculations are performed. By letter dated May 6, 1997, submitted on the GGNS docket, the licensee requested NRC review and acceptance of the results of the BWROG/GE analysis report (Reference 3) for replacement of the 30-second minimum time for fission product release specified in NUREG-1465. This analysis, performed by GE, calculates the minimum time to the onset of gap activity release to be 121 seconds. The GE analysis uses a bounding BWR plant configuration and reactor fuel design, and is intended to be generically applicable to all currently operating BWR plants using currently licensed BWR fuel. The BWROG/GE report was reviewed and accepted by the staff by letter dated September 9, 1999 (Reference 4), as applicable to all BWR plants.

The licensee's application requests plant specific approval of the implementation of the results of Reference 3 for revision of the GGNS licensing basis.

3.0 EVALUATION

Applicable portions of the staff's safety evaluation, provided in Reference 4 for review and acceptance of the generic BWROG/GE report, are repeated below to facilitate the technical justification of the specific implementation of the results for GGNS.

GE performed its calculations using loss-of-coolant accident (LOCA) methodology that had been previously reviewed and approved by the staff. In this methodology, the SAFER computer code calculates the long-term system response of the reactor. The CHASTE computer code is then used to model fuel rod heatup for the highest power axial plane in the highest power fuel rod assembly. GE used this approved methodology to evaluate the minimum time to fuel cladding perforation, and to perform sensitivity studies in determining the most limiting BWR vessel design, fuel rod design, and core burnup.

As further verification of the minimum time to the onset of gap activity release following a postulated design-basis LOCA in the BWROG report, the NRC technical assistance contractor, Idaho National Engineering & Environmental Laboratory (INEEL), performed a confirmatory calculation for NRC using the SCDAP/RELAP5 computer code for thermal-hydraulic calculations and the FRAPCON3 computer code for fuel rod failure calculations. INEEL prepared a technical evaluation report for NRC, "Evaluation of Fuel Pin Failure Timing in Boiling Water Reactors," dated July 1999 (Reference 5).

The staff has also performed an independent and specific confirmatory calculation for GGNS using a series of LOCA analyses to evaluate the minimum time to the onset of fission product release from perforated fuel rods following a postulated design-basis LOCA. The staff used its TRAC-BF1 best estimate system code with a GGNS input model provided by INEEL and the FRAPCON3 code to estimate the fuel initial conditions.

The purpose of the staff analyses was to confirm that the GE analyses in the BWROG report are conservative, and that the revision to the GGNS fuel failure assumptions requested by the licensee are acceptable.

3.1 Design-Basis Accident

During promulgation and development of NUREG-1465, the staff conducted a review of current plant final safety analysis reports to identify all design-basis accidents in which the licensee had identified fuel failure. For all accidents with the potential for release of fission products, the class of accidents that had the shortest time until the first fuel rod failure was the design-basis LOCA with complete emergency core cooling system (ECCS) failure. Therefore, the staff concluded that a postulated large break LOCA with complete ECCS failure was a reasonable initiator for modeling the earliest appearance of the fuel gap activity (i.e., minimum time to the onset of fission product release from a perforated fuel rod).

3.2 Reactor Fuel Design

The most limiting fuel design for evaluation of earliest fuel rod perforation would include the highest peak linear heat generation rate (PLHGR), the highest stored energy, and the highest internal pressure. To determine the most limiting fuel design, GE evaluated the following different BWR fuel designs that are currently in use: GE8, GE9B, GE10, GE11, and GE12, and Siemens 8 x 8 and 9 x 9 fuels to determine the most limiting fuel design. In its evaluation of earliest fuel rod perforation, GE considered two major parameters, PLHGR and fuel rod internal pressure.

Using the PLHGR as an input value, GE evaluated the sensitivity of peak cladding temperature (PCT) to different lattice designs, fuel exposures, and vessel designs. These sensitivity studies were necessary to evaluate the impact of such important parameters as depressurization rate, fill gas pressure, radiation heat transfer, and pin power distribution. On the basis of these sensitivity studies, GE determined that GE11 fuel in a 205-inch inside diameter (ID) vessel with a 28-inch ID recirculation suction line (see Section 3.3) would be the most limiting case. The staff reviewed the sensitivity studies submitted by GE and concurs with GE's conclusion.

3.3 Primary Coolant System Design

The most limiting primary coolant system design for evaluation of earliest fuel rod perforation would be the combination of the smallest reactor pressure vessel (RPV) water inventory with the largest primary coolant break flow rate. The combination of these two parameters will lead to the fastest RPV water inventory depletion resulting in earliest RPV core uncovery. GE used geometry ratios for various BWR designs to select the most limiting primary coolant system design. The BWR design with the smallest ratio (i.e., RPV inventory to break flow rate) will have the earliest core uncovery time.

GE determined that the BWR-4 design, with an RPV ID of 205 inches and a 28-inch ID recirculation pipe, would be the most limiting primary coolant system design for evaluating the earliest fuel rod perforation. The staff accepted GE's determination that Vermont Yankee (VY), with a 205-inch RPV ID and a 28-inch ID recirculation pipe, has the same limiting primary coolant system design. Therefore, INEEL performed a confirmatory calculation using VY plant data to check the GE analysis for evaluating the earliest fuel rod perforation.

3.4 Accident Sequence Models

The thermal and hydraulic design characteristics of the core and nuclear fuel data used by the staff, INEEL, and GE are as follows:

Parameter	VY (INEEL)	GGNS (NRC)	GE/BWROG (GE)
Thermal design output, MWt	1593	3833	1880
RPV diameter, inches	205	251	205
Recirculation pipe diameter, inches	28	22/24	28
Number of fuel assemblies	368	800	484
Fuel configuration	8 x 8	9 x 9	9 x 9

3.4.1 Confirmatory Analysis by INEEL

INEEL converted the VY RELAP5/MOD3 computer code input deck to the current version of the SCADAP/RELAP5 code. The SCDAP code models the reactor core behavior during a postulated reactor accident, and the RELAP5 code calculates the overall reactor coolant system thermal hydraulics. The analytical methodologies that used the SCDAP/RELAP5 code and the FRAPCON3 code are given in detail in Reference 5. The VY reactor core consists of a total of 368 fuel assemblies with thermal design output of 1593 MWt. Each VY fuel assembly is configured in an 8 x 8 fuel rod array with one large centrally located water rod that occupies the space that would otherwise accommodate four fuel rods. GE's limiting core design consists of a total of 484 fuel assemblies with thermal design output of 1880 MWt. Each GE11 fuel assembly is configured in a 9 x 9 fuel rod array with two large centrally located water rods that occupy the space that would otherwise accommodate seven fuel rods.

INEEL calculated two design-basis LOCA transients for the near beginning of life (BOL) and at the end of life (EOL) using the SCDAP/RELAP5 code with the corresponding FRAPCON3 code core temperature to determine the minimum time from reactor accident initiation to the first fuel rod perforation. The results indicated that the high power and high stored energy conditions near the BOL will lead to an earlier fuel rod failure than high fuel rod pressure at the EOL. INEEL calculated 152 seconds for the minimum time to onset of gap activity release for the conditions near the BOL. The staff finds that the value of 152 seconds calculated by INEEL indicates that the 121 seconds calculated by GE in the BWROG report is a conservative value.

3.4.2 NRC Staff Confirmatory Analysis for GGNS

The staff obtained a GGNS input deck originally prepared by INEEL. This deck was written for a previous version of the TRAC computer code and, therefore, had to be modified to run with the BF1 version of TRAC. All of the plant geometry was retained with the exception of the jet pumps. The staff used a jet pump model, which was extracted from another BWR/6 input deck, and changed the number of jet pumps to correspond to the number at GGNS. The deck consists of 46 components. The major parameters used in the deck are given in Table 1. The staff used information from the most recent core operating limit's report dated August 13, 1997 (Reference 6), to determine the fuel type being used at GGNS, and used the staff's lattice physics methods to predict radial peaking factors for use in the CHAN component type. The part length fuel rods were not modeled and the internal water channels were modeled with water pins. The axial power distribution is shown in Figure 1 and the remaining kinetics parameters were taken to be the default values.

The deck was tested preceding its use by running it to steady state. The steady-state condition is summarized in Table 2. The results show that the core exit quality is slightly higher than the plant design and the feedwater temperature was set lower than design to achieve 10 degrees Kelvin (K) of subcooling at the channel inlet. These differences were not entirely unexpected, given the fact that the steam separators were modeled using the simple separator option in the vessel component. These differences are considered acceptable.

The TRAC deck was run for the design-basis recirculation suction line break with no emergency core cooling injection and the fuel was allowed to heat up. This scenario was run for several different linear heat generation rates, recirculation line sizes, and pin power distributions, and used two different rod groupings to evaluate some of the potential sensitivities and to ensure that the analysis maximized the heatup rate. Fuel pin internal pressure was assumed to be the value at the time of peak reactivity (which was assumed to be the time of peak power) and was predicted using the FRAPCON-3 code. The failure temperature was estimated to be 1000 degrees K using the methods and data in NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Reference 7).

The PCT as a function of time is presented in Figure 2. As shown in the figure, the minimum time to failure is estimated to be 172 seconds. In order to confirm that the model is behaving as expected, the staff also examined the steam dome pressure (Figure 3) and the break flows (Figure 4), and confirmed that the ECCS injection flows were zero by examining the computer output file.

3.4.3 Evaluation Conclusions

The calculated minimum times to the onset of gap activity release following a postulated design-basis accident for the analytical models considered are as follows:

GE calculation in BWROG report:	121 seconds
INEEL calculation using VY design:	152 seconds
NRC calculation for GGNS:	172 seconds

The plant-specific calculation results for GGNS demonstrate a conservative upper bound for the minimum time to the onset of gap activity release. On the basis of this evaluation, the licensee's request to revise the GGNS UFSAR to change the minimum time to the onset of fission product release from perforated fuel rods following a postulated design-basis accident to 121 seconds from the currently specified instantaneous release is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 67333, December 1, 1999). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: Tables 1 and 2
Figures 1 through 4

Principal Contributors: S. Patrick Sekerak
Jay Y. Lee

Date: March 22, 2000

REFERENCES

1. U.S. Nuclear Regulatory Commission, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.

2. U.S. Atomic Energy Commission, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962.

3. General Electric Company Report, "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR," July 1996.

4. U.S. Nuclear Regulatory Commission letter, S. P. Sekerak to W. A. Eaton, Entergy Operations, Inc., "Acceptance of BWROG Report "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR," dated July 1996," dated September 9, 1999.

5. D. L. Knudson and R. R. Schultz, Idaho National Engineering and Environmental Laboratory, "Evaluation of Fuel Pin Failure Timing in Boiling Water Reactors," August 1999.

6. Letter from J. J. Hagan (Entergy) to USNRC, "Core Operating Limits Report (GGNS-MS-48.0, Revision 5) for Cycle 9 Submittal," August 13, 1997.

7. D. A. Powers and R. O. Meyer, U.S. Nuclear Regulatory Commission, NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," April 1980.

Table 1 TRAC-BF1 Input Deck Description

Component Type	Number	Notarization
VESSEL	1	12 Axial, 4 Radial, 2 Theta
CHAN	8	27 Axial, 7 or 15 Fuel Rod Groups
JETP	2	Standard
SEPD	0	Used VESSEL Perfect Separator
Recirc Loops	2	22 Nodes Total
Steam Lines	2	10 Nodes Total
Feedwater	2	0 (Used FILL Directly Connected to VESS)

Table 2

Grand Gulf Model Steady-State Results

Parameter	Value
Total Power	3833 MW_t
Core Flow	14,630 kg/sec (1.16×10^8 lb/hr)
Steam Flow	2000 kg/sec (1.59×10^7 lb/hr)
Core Exit Quality	15.7 %
Feedwater Temperature	460 K (368 °F)

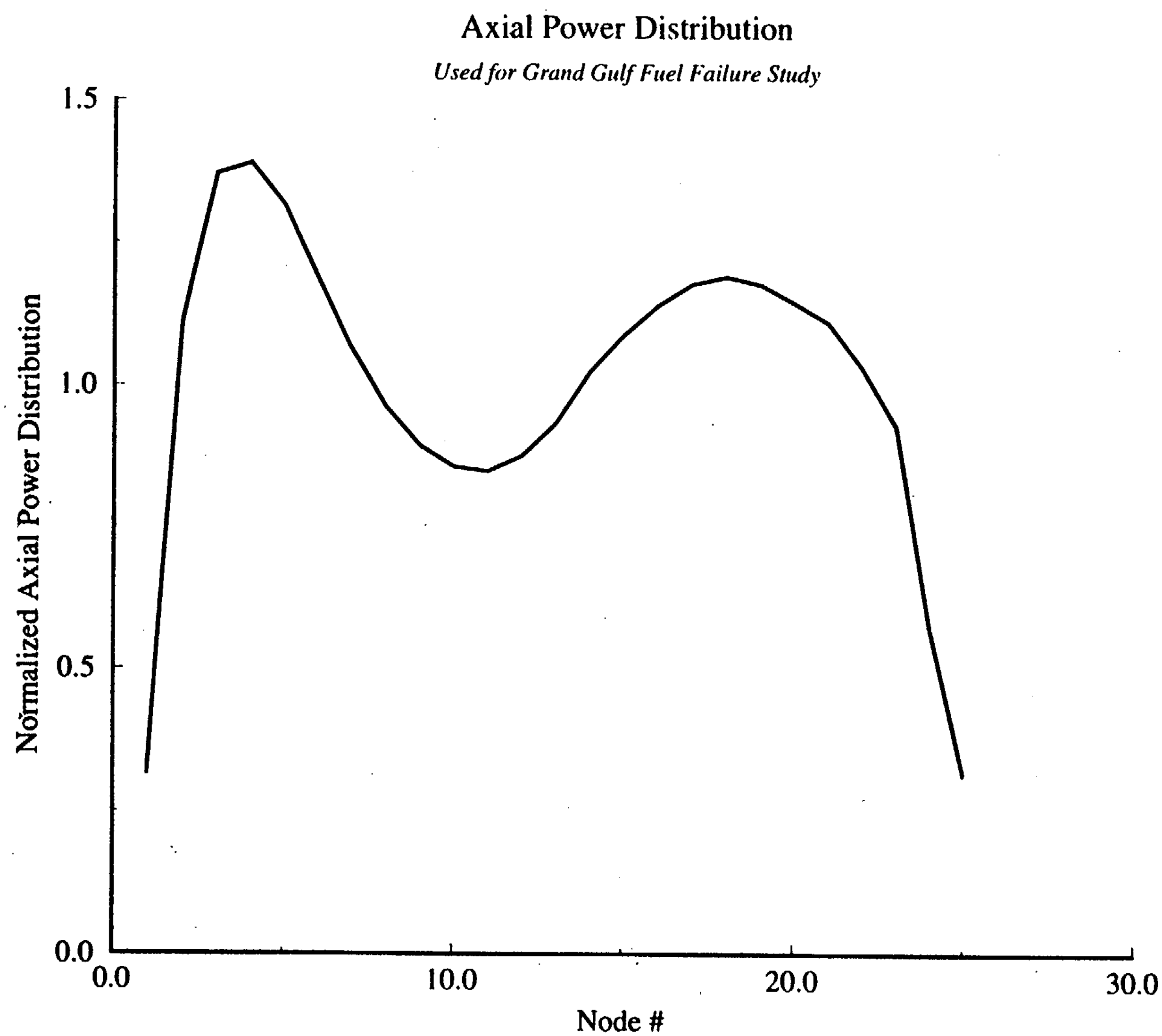


Figure 1 Core Average Axial Power Distribution Used in TRAC-BF1 Analysis

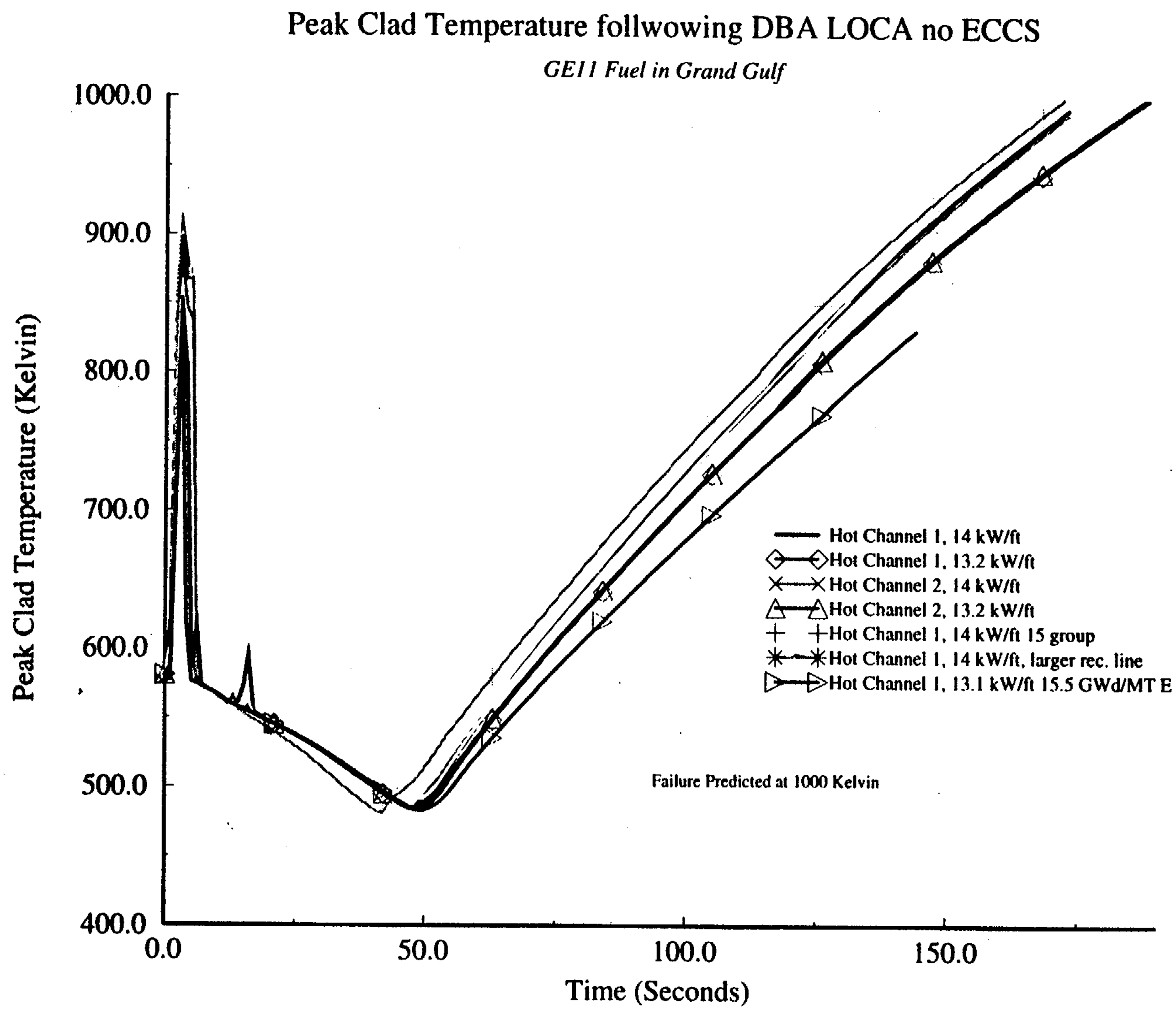


Figure 2 TRAC-BF1 Predicted Peak Cladding Temperatures for Grand Gulf Design Basis LOCA with no ECC Injection

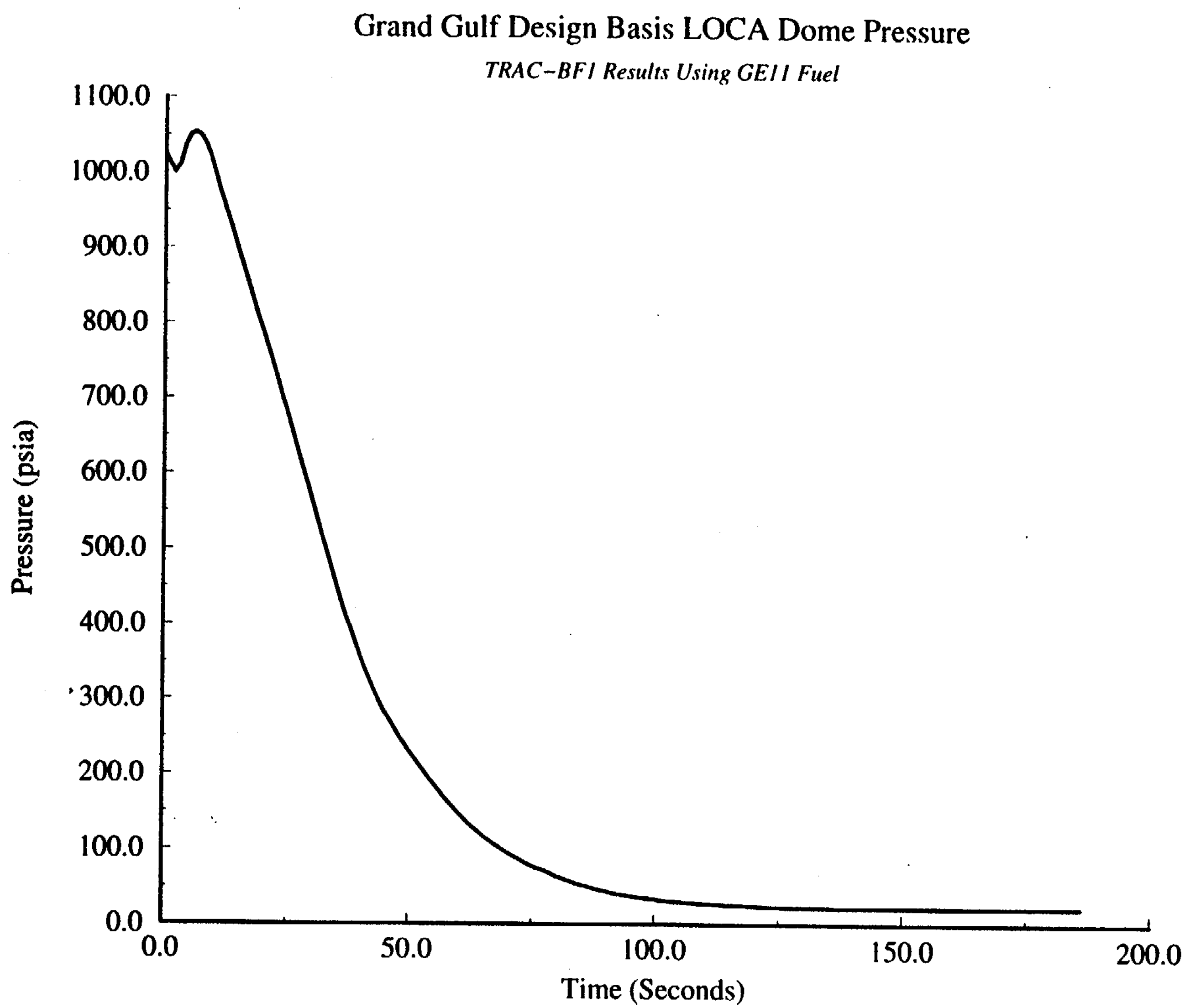
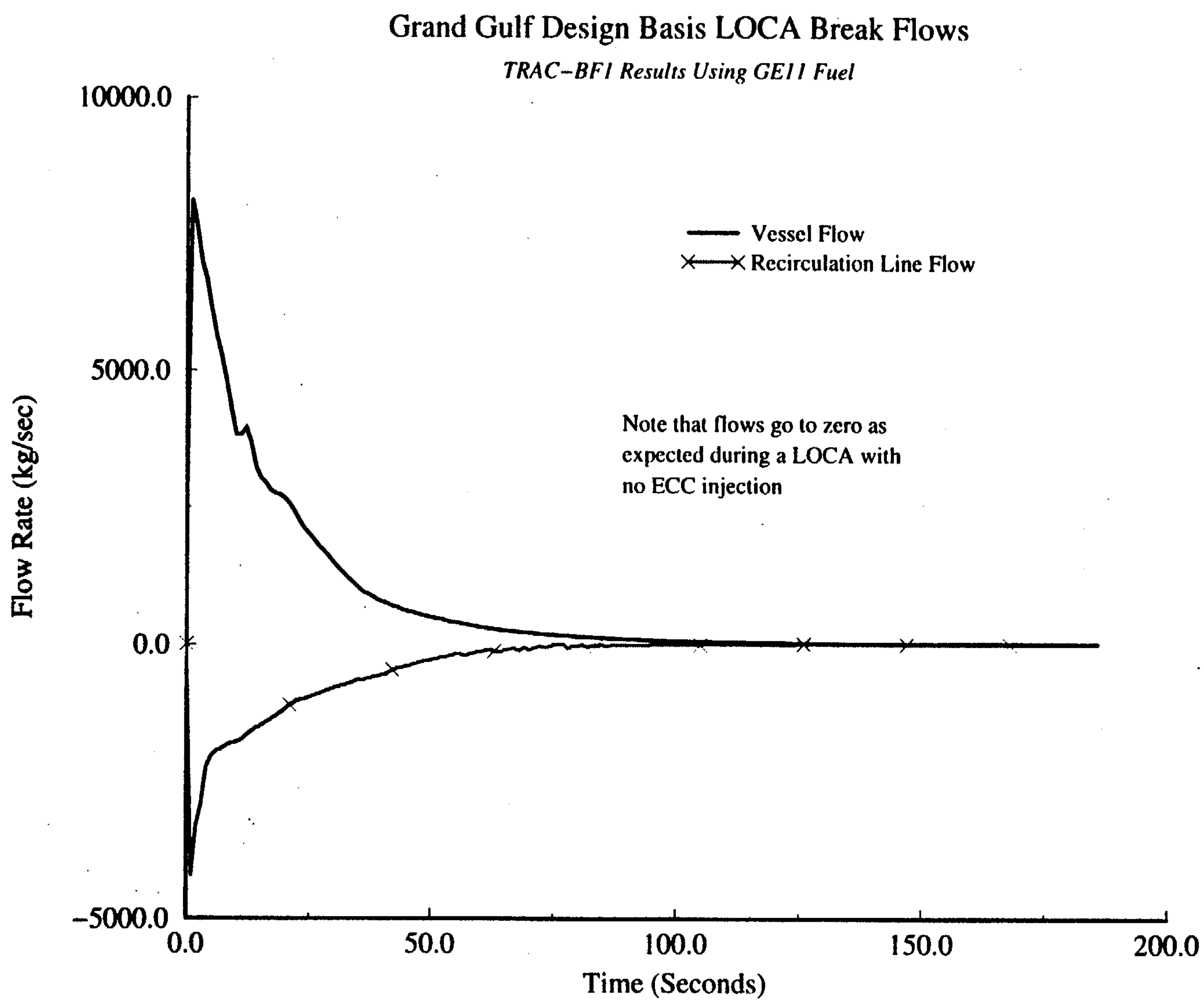


Figure 3 TRAC-BF1 Predicted Steam Dome Pressure During Design Basis LOCA with no ECC Injection



Figrue 4 TRAC-BF1 Predicted Break Flows During Design Basis LOCA with no ECC Injection