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September 4, 1998

Docket Nos. 50-321
50-366

HL-5674

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant
Request for Additional Information
Extended Power Uprate License Amendment Request

Gentlemen:

As discussed in the Nuclear Regulatory Commission (NRC) Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting on August 26, 1998, this letter provides information related to the Edwin I. Hatch Nuclear Plant Probabilistic Risk Assessment review. The enclosure contains a summary description of the presentation given by Southern Nuclear Operating Company in the meeting.

Should you have any questions in this regard, please contact this office.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

DLM/ld

Enclosure: Response to Request for Additional Information on the Extended Power Uprate License Amendment Request.

cc: See next page.

A001

cc: *Southern Nuclear Operating Company*
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

Enclosure

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1.0 Background

An evaluation of the Edwin I. Hatch Nuclear Plant probabilistic risk assessment (PRA) was performed to support the license amendment for extended power uprate. Although Plant Hatch is not committed to Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) 19 or Regulatory Guide 1.174, the PRA evaluation was supplemented to include an uncertainty analysis performed to meet the intent of this guidance. The supplemental analysis results were presented in the NRC Advisory Committee on Reactor Safeguards (ACRS) Thermal-Hydraulic Phenomena subcommittee meeting on August 27, 1998. At the request of the NRC staff, Southern Nuclear Operating Company (SNC) provides the following summary description of the analysis.

2.0 PRA Quality

The Plant Hatch PRA was developed in response to NRC Generic Letter 88-20. The PRA was developed by engineering and licensing personnel with extensive experience in nuclear plant design and operation. Calculations prepared for the PRA were developed and are maintained as quality records, which are retained for the life of the plant. An independent review group and General Electric personnel performed a peer review of the assessment.

The PRA evaluation performed to support extended power uprate has similar quality attributes. Engineering and licensing personnel familiar with the original PRA performed the evaluation, and the calculations were developed and are maintained as quality records.

3.0 Uncertainty Assessment

Each of the following NRC Regulatory Guide 1.174 classes of uncertainty is addressed in the supplemental analysis performed for the Plant Hatch extended power uprate:

- Parameter uncertainty
- Model uncertainty
- Completeness uncertainty

The intent is to demonstrate that the extended power uprate changes do not have significant impacts on the PRA uncertainty, and do not result in unique risks or vulnerabilities.

3.1 Parameter Uncertainty

Using the Unit 1 internal events PRA and RISKMAN software, typical parametric uncertainty analyses were performed for the original licensed power level and the extended uprate power level. The uncertainty analysis methodology uses a Monte Carlo sampling scheme so that the potential impacts from knowledge state dependencies can be included in the determination of core damage frequency (CDF) and large early release frequency (LERF). As an example, since the same failure data is used for all pumps in a system, there is a dependency among the pump failure rates. The sampling scheme is designed to incorporate these dependencies in the calculation of CDF and LERF. In addition to these failure data dependencies, the dependencies among split fractions and the top events of the sequence event trees were given a conservative treatment of correlation. For example, loss of onsite and offsite power, and potential power recovery were correlated. HPCI and RCIC failures were also correlated, as were the loss of feedwater initiating event and potential condensate recovery. For the extended power uprate case, the failure of the operator depressurization action in medium LOCA sequences (DE4) used the new success likelihood index method (SLIM) estimate of 8E-02.

The results of the uncertainty assessments are given in Tables 1 and 2, and Figures 1-4. The shapes of the uncertainty distributions are virtually unchanged between the original power level case and the extended power uprate case. As expected, they are simply shifted to reflect the small increase in mean CDF and LERF. Tables 1 and 2 provide the mean, median, 95% and 5% upper and lower bounds, and the approximate error factor. The error factor is a measure of the spread of a distribution from the median to the 95% upper bound, generally taken to be lognormally distributed. It was calculated either as the 95% upper bound divided by the median, or as the square root of the 95% upper bound divided by the 5% lower bound. These equivalent calculations provide approximately the same error factor, depending on the closeness of fit of the distributions to the lognormal. As shown in Tables 1 and 2, the error factors for the original power level case and the extended power uprate case are about a factor of 2.3 for CDF, and about 4 for LERF. These are very reasonable ranges, and indicate that the parametric uncertainty is fairly low.

3.2 Modeling Uncertainty

Modeling uncertainty is associated with the basic ground rules, assumptions, and decisions made when developing the PRA fault tree and event tree models, and the MAAP thermal-hydraulic model. They are usually addressed using sensitivity studies and qualitative assessments. For the Plant Hatch PRA, a full range of sensitivity studies was performed for the IPE, including extensive MAAP parameter sensitivity studies. These

sensitivity studies demonstrate that the major modeling uncertainties do not significantly alter the conclusions of the risk assessment, or cause risk vulnerabilities.
For the extended power uprate, sensitivity studies were performed for various operator actions, and additional MAAP deterministic assessments were made to examine timing and release category groupings. As discussed qualitatively, the extended power uprate changes primarily affect timing of operator actions, and do not affect the types of initiators, sequences, success criteria, systems models, or plant configuration and operation. Therefore, model uncertainty is not significantly impacted by the power uprate.

3.3 Completeness Uncertainty

Completeness uncertainty arises from limitations in the scope of the PRA. In the case of the Plant Hatch extended power uprate, assessment of the internal events risk and the fire risk was performed in a quantitative evaluation. However, Plant Hatch had previously performed a seismic margins assessment rather than a seismic PRA, and Plant Hatch does not have a shutdown risk assessment. In order to provide better information of the potential risk and impacts, a simple seismic sensitivity assessment was performed, and a qualitative assessment of shutdown risk impacts was made following the intent of the guidance in the new Standard Review Plan (SRP) 19.

3.3.1 Seismic Sensitivity Study

The Plant Hatch Seismic Margins assessment demonstrated that all safe shutdown equipment list (SSEL) components, including containment performance components, screen at 0.3g HCLPF. HCLPF is defined as "high confidence of a low probability of failure." Using the 0.3g HCLPF, a median capacity of 0.79g was calculated, with combined beta uncertainty factor of 0.5. LLNL and EPRI mean seismic hazard curves were convolved with this seismic fragility for "surrogate" components. It was conservatively assumed that these "surrogate" components were singletons that lead to core damage. The results are given in Table 3.

This sensitivity is not a seismic PRA, but does indicate seismic CDF would likely be in 1E-6 to 1E-5/year range, depending on which seismic hazard curve is used. Therefore, this additional seismic CDF would not revise the appropriate RG 1.174 region. In addition, extended power uprate does not directly affect seismic models, although the operator actions could be affected similar to the internal events PRA .

3.3.2 Shutdown Risk

A Plant Hatch shutdown risk model is not available, so a quantitative analysis could not be performed. Other shutdown PSAs were reviewed, and showed shutdown CDF generally less than or equal to "at power" CDF, but there are exceptions, and the scope of the shutdown PSAs varied widely. Therefore, shutdown risk impacts were examined in a qualitative manner.

Higher power levels would primarily impact the initial cooldown interval, although there would be some small impacts during the later phases of shutdown as well. SRP 19 posed four questions to determine if impacts on shutdown risk would be important:

a. Will these changes affect shutdown schedule?

   Although it would take a few hours longer to reduce temperature and pressure to closed cycle RHR entry conditions, and to achieve cold shutdown, there would be very little change to the shutdown schedule, and no direct safety impacts on the schedule.

b. Will these changes affect operator ability to respond?

   As with the internal events "at power" PRA, the increase in decay heat will result in a small decrease in the time available for operator actions during shutdown. However, operator action times during shutdown are much longer relative to the times for "at power" operator actions. For example, during cold shutdown the operators have many hours, or days, to recover core cooling, so any small decrease in time for operator action would not be a critical factor in the human error probability (HEP).

c. Will changes affect shutdown equipment reliability?

   Equipment reliability will be maintained to the current standards due to the fact that equipment will be operated within acceptable design and operational limits. No impacts on shutdown equipment reliability are expected due to the extended power uprate.

d. Will changes affect availability of equipment or instrumentation used for contingency plans?

   No impacts on equipment availability or instrumentation used for contingency plans are expected due to the extended power uprate.

Based on these qualitative responses to SRP 19, extended power uprate has no unique or significant impacts on shutdown risk.

4.0 Summary and Conclusions

Using the general guidance in RG 1.174 and SRP 19, the parametric, modeling, and completeness uncertainties were assessed. The extended power uprate changes do not have a significant impact on the perceived uncertainties, and do not pose unique uncertainties or risk vulnerabilities.

Table 1
Parameter Uncertainty-CDF
Unit 1 Internal Events

CDF	MEAN	MEDIAN	95%	5%	"EF"
Original Power	2.5E-05	2.1E-05	4.7E-05	1.2E-05	2.3
Power Uprate	2.6E-05	2.2E-05	4.9E-05	1.2E-05	2.3

Table 2
Parameter Uncertainty-LERF
Unit 1 Internal Events

LERF	MEAN	MEDIAN	95%	5%	"EF"
Original Power	5.2E-06	3.1E-06	1.5E-05	9.2E-07	4.0
Power Uprate	5.5E-06	3.0E-06	1.6E-05	9.3E-07	4.1

Table 3
Seismic Sensitivity Analysis
Core Damage Frequency Estimate

SENSITIVITY CASE	EPRI Hazard	LLNL Hazard
1 Surrogate Failure	8.4E-07	5.0E-06
3 Surrogate Failures	2.0E-06	1.0E-05

Figure 1
Parametric Uncertainties-Core Damage Frequency
Internal Events-Unit 1

Original Licensed Power Level

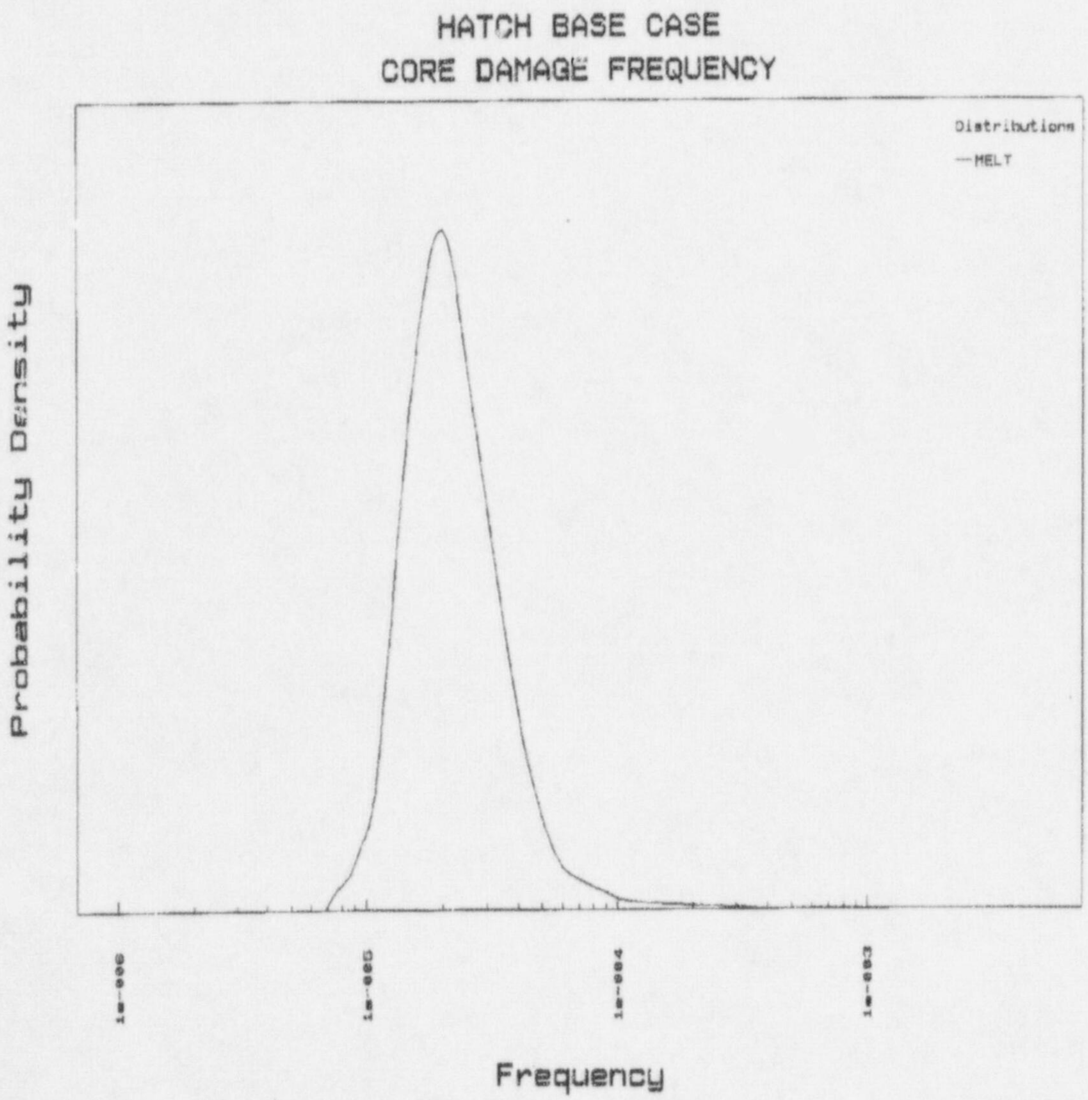

Figure 2
Parametric Uncertainties-Core Damage Frequency
Internal Events-Unit 1

Extended Uprate Power Level

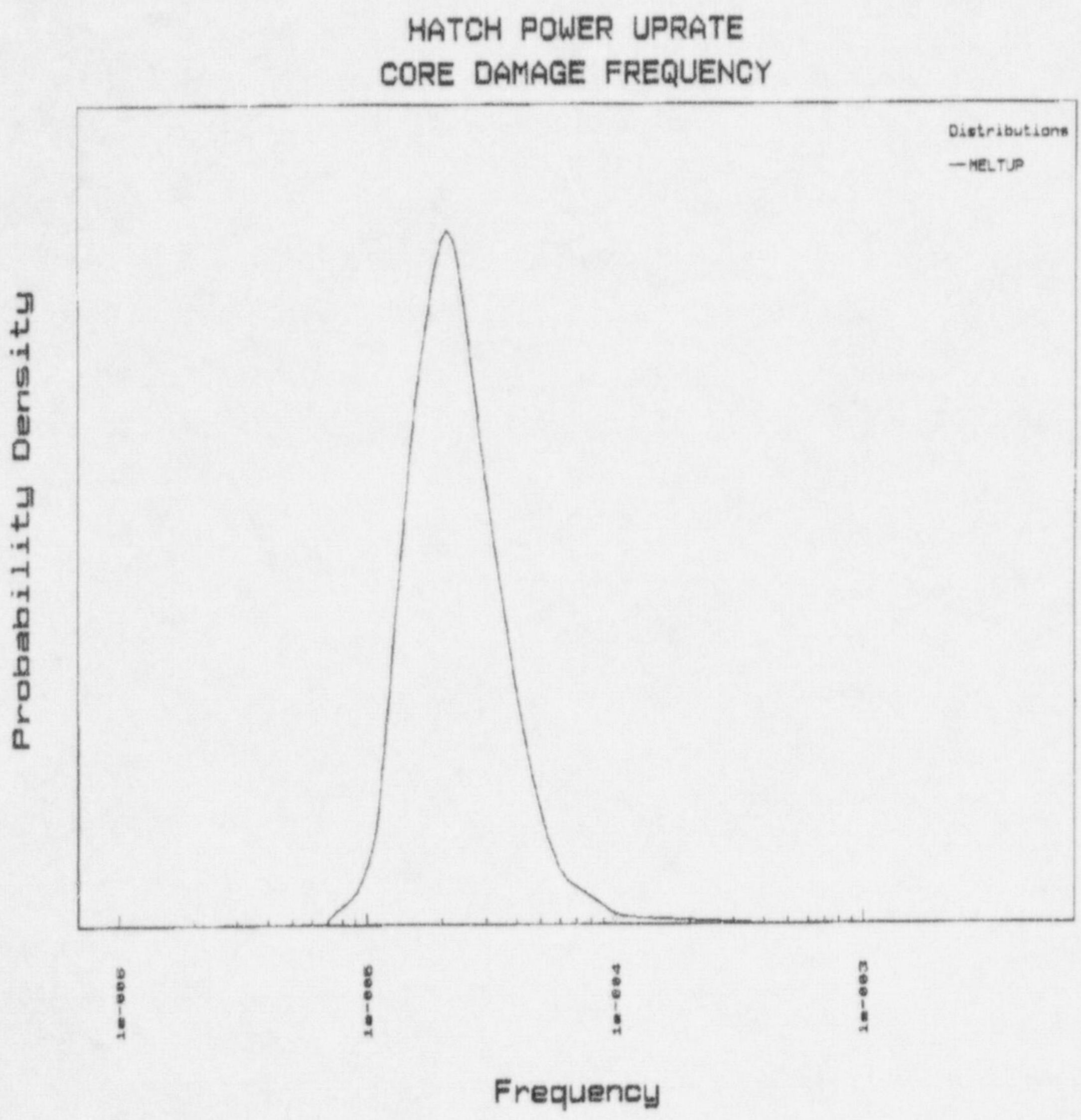

Figure 3
Parametric Uncertainties-Large Early Release Frequency
Internal Events-Unit 1

Original Licensed Power Level

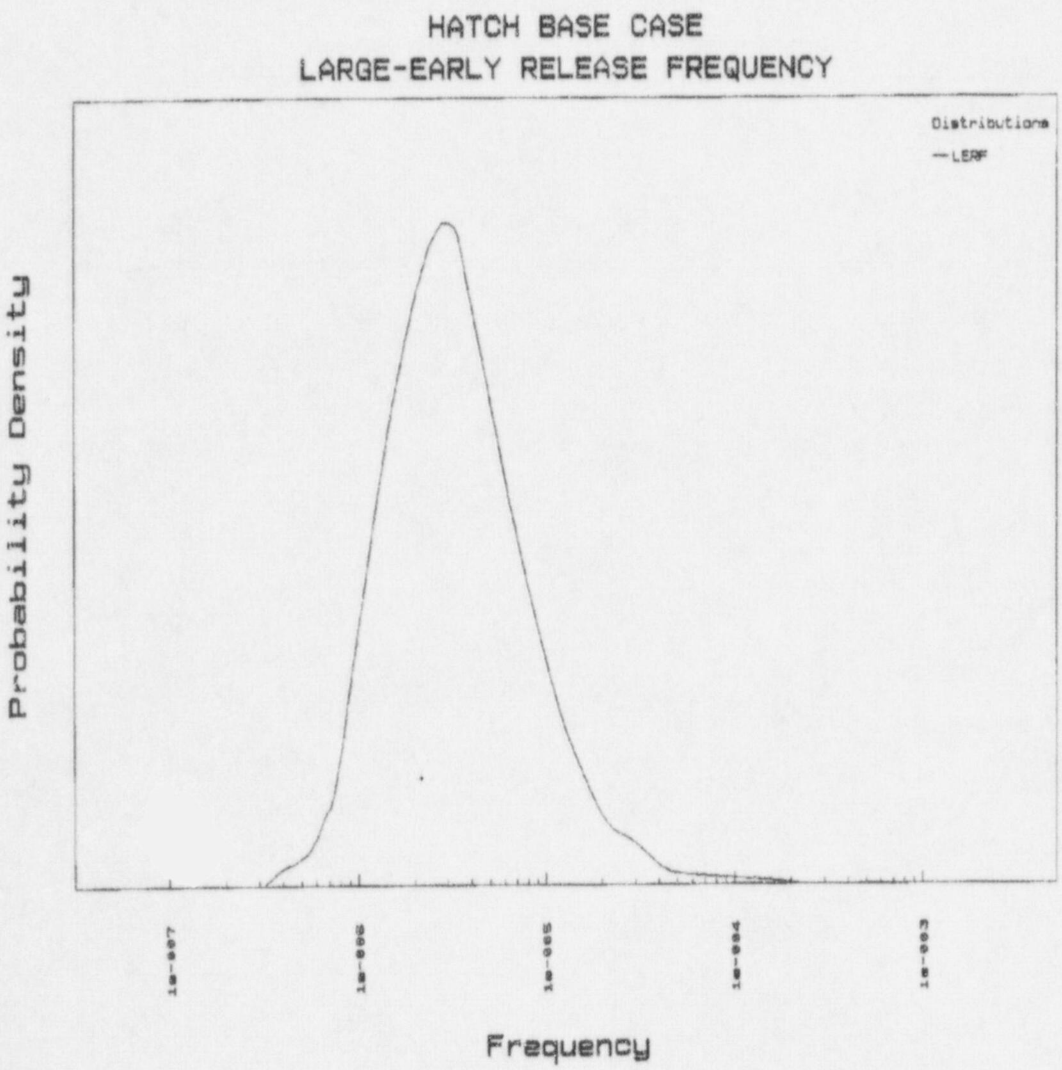

Figure 4
Parametric Uncertainties-Large Early Rclease Frequency
Internal Events-Unit 1

Extended Uprate Power Level