Entergy

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William A. Eaton
Vice President,
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Grand Gulf Nuclear Station

GNRO-2002/00020

March 1, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Grand Gulf Nuclear Station, Unit 1
Docket 50-416
Supplemental Information for License Amendment Request
Removal of Operating MODE Restrictions for Performing
Emergency Diesel Generator Testing (LBDC-2002/003)

REFERENCES: Letter from W. A. Eaton to USNRC, "Proposed Amendment of Facility Operating License to Remove MODE Restrictions for Performing Emergency Diesel Generator Testing" dated November 15, 2001.

Dear Sir or Madam:

By the letter referenced above Entergy Operations, Inc. (Entergy) proposed a change to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specification (TS). The letter dealt with removing MODE restrictions from certain surveillances on the Emergency diesel Generators.

On January 16, 2002 members of your staff and personnel from GGNS conducted a telephone conference to discuss the proposed change. In particular section 4.7 "Risk Assessment" was discussed. During the discussion it became apparent that a supplemental response was needed to clarify the information provided in the original submittal. Entergy's supplemental information is provided in attachment 1.

There are no Technical Specification changes proposed by this supplemental response. The original no significant hazards considerations included in the reference is still valid and is not affected by any information contained in the supplement. There are no new commitments contained in this letter.

Entergy still requests approval of the proposed amendment by August 01, 2002 in order to allow for work planning prior to the fall refueling outage. Once approved, the amendment shall be implemented within 60 days. Although this request is neither exigent nor emergency, your prompt review is requested.

A001

If you have any questions or require additional information, please contact Lonnie F. Daughtery at extension (601) 437-2334.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 1, 2002.

Sincerely,

WAE/LFD

Attachment:
Supplemental Risk Assessment Information

cc: Mr. Ellis W. Merschoff
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
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Mr. S. P. Sekerak, NRR/DLPM (w/2)
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Mr. T. L. Hoeg, GGNS Senior Resident
Mr. D. E. Levanway (Wise Carter)
Mr. L. J. Smith (Wise Carter)
Mr. N. S. Reynolds
Mr. H. L. Thomas

March 1, 2002
GNRO-2002/00020

bcc:
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OTHER: File (LRS_DOCS Directory - GNRI or GNRO)

Dr. E. F. Thompson (w/a)
State Health Officer
State Board of Health
P. O. Box 1700
Jackson, Mississippi 39205

Attachment 1 to GNRO-2002/00020

4.7 Risk Assessment

During certain portions of the surveillances the DG would not be able to immediately respond to an accident. DG unavailability during the performance of the proposed on-line DG testing is summarized in Attachment 4, with the longest unavailability time of 8.0 hours. Based on these estimates, the increase in average Core Damage Frequency (CDF) and Incremental Conditional Core Damage Probability (ICCDP) is determined as follows:

For the average maintenance model (as specified in RG 1.177), the base core damage frequency for GGNS is 5.46E-6 per year. Conservative estimates of the equivalent yearly core damage probability when a DG is out of service (for the whole year) can be made utilizing the risk achievement worth for each of the DGs. This results in the following CDF estimates:

	CDF on Yearly Basis
Baseline	5.46E-6
DG A OOS	1.19E-5
DG B OOS	1.72E-5
DG C OOS	2.90E-5

Average CDF Increase

The average at-power CDF with the additional out of service time for all three DGs is computed by adding the CDF for the additional period during which the DG is out of service with the CDF for the remainder of the year. The change in CDF is calculated as follows:

$$\Delta CDF_{At-Power} = \frac{T_A}{T_{Year}}(CDF_{AOOS}) + \frac{T_B}{T_{Year}}(CDF_{BOOS}) + \frac{T_C}{T_{Year}}(CDF_{COOS}) + \left(1 - \frac{(T_A + T_B + T_C)}{T_{Year}}\right)(CDF_{Base}) - CDF_{Base}$$

where,

CDF_{AOOS}, CDF_{BOOS}, and CDF_{COOS} are the estimated yearly CDF with the corresponding DG out of service.

T_A, T_B, and T_C are the additional out of service times for each DG due to the proposed on line testing. This is estimated to be a total of 12 hours per cycle for each diesel. On a yearly basis this number is 8 hours per diesel per year with the assumption of an 18 month cycle.

T_{Year} is the number of hours in a year (8760 hours).

CDF_{Base} is the baseline annual average CDF with the current average unavailability of the DGs.

Therefore, the ΔCDF associated with this change is:

$$\Delta CDF_{At-Power} = \frac{8\,hrs}{8760\,hrs}(1.19E-5/yr) + \frac{8\,hrs}{8760\,hrs}(1.72E-5/yr) + \frac{8\,hrs}{8760\,hrs}(2.9E-5/yr) + \left(1 - \frac{(8hrs + 8\,hrs + 8\,hrs)}{8760\,hrs}\right)(5.46E-6/yr) - 5.46E-6/yr$$
$$= 3.81E-8/yr$$

This value for ΔCDF is significantly smaller than the RG 1.174 guidance of less than 1.0E-6/year for very small CDF increases.

ICCDP

The incremental conditional core damage probability (ICCDP) can be computed using the definition in RG 1.177. In terms of the above defined parameters, the definition of ICCDP associated with the Division 3 (DG C) out of service is as follows:

$$ICCDP_C = \frac{T_C}{8760\,hrs/yr}(CDF_{COOS} - CDF_{Base})$$

There is a similar expression for each of the other DGs, but since the CDF for DG C bounds that of the other 2 DGs, it is used for the calculation of ICCDP. The total increase on out of service time (12 hours) is also used for this calculation.

$$ICCDP = \frac{12\,hrs}{8760\,hrs/yr}(2.9E-5/yr - 5.46E-6/yr)$$
$$= 3.22E-8$$

This value for ICCDP is significantly smaller than the RG 1.177 guidance of 5.0E-7 for a small quantitative impact.

ΔLERF and ICLERP

Calculation of ΔLERF and ICLERP are not necessary as these two are a fraction of ΔCDF and ICCDP and both ΔCDF and ICCDP are below the respective ΔLERF and ICLERP significance guidance from RG 1.174 and RG 1.177.

PSA Quality

The original GGNS Individual Plant Examination (IPE) was developed by Entergy with the assistance of Science Applications International Corporation (SAIC) and was submitted to the NRC in 1992. It was revised in 1997 and was renamed the GGNS PSA, revision 1. The above evaluations were performed using results from the Revision 1 GGNS PSA. This revision of the PSA is currently undergoing a major revision but results are not yet available. However, an independent assessment of the Revision 1 GGNS PSA has been completed to ensure that the GGNS PSA was comparable to other PSA programs in use throughout the industry. This assessment applied the Self-Assessment Process developed as part of the Boiling Water Reactor Owners' Group (BWROG) PSA Peer Review Certification Program. The PSA Certification Team, which was a group of Industry and Utility experts selected by the BWROG, completed an inspection and review of the GGNS PSA in August 1997 and completed a PSA Certification Report in November 1997. The models and methodology used in Revision 1 of the GGNS PSA were included in the PSA Certification review. The quality of the PSA and completeness of the PSA documentation were also assessed. The certification team found that the GGNS PSA is fully capable of addressing issues requiring risk significance determination with a few enhancements. Because the proposed changes to the GGNS Facility Operating License for online diesel generator testing have only a small impact on total DG unavailability, any enhancements made to the GGNS PSA are not expected to significantly impact the overall conclusions of the above evaluations.

External Events

By letter dated November 15, 1995, Entergy Operations, Inc. (EOI) submitted the Individual Plant Examination for External Events (IPEEE) for GGNS. In the IPEEE, seismic was addressed using a seismic margins methodology, fire was addressed using fire PRA methods (i.e., EPRI TR-105928, Fire PRA Implementation Guide), and the other events were addressed by demonstrating conformance to the 1975 SRP. EOI received the NRC Staff Evaluation Report by letter dated March 16, 2001, in which the staff concluded that the aspects of seismic events, fires and high winds, floods and other (HFO) events were adequately addressed. Of the considered events, fire and seismic are initiators with the most potential for an induced loss of offsite power. A loss of offsite power is relevant to the proposed changes because of the potential increase in DG unavailability.

GGNS was classified in NUREG-1407 as a reduced scope plant of low seismicity and emphasis was placed on conducting seismic walkdowns for the IPEEE. Therefore, a seismic loss of offsite power (LOOP) initiator frequency was not determined but can be estimated as follows. Ceramic insulators for offsite power transformers tend to be the most vulnerable components in the offsite power system during a seismic event. NUREG/CR-4550, Vol.4, Rev. 1, Part 3, "Analysis of Core Damage Frequency, Peach Bottom Unit 2," estimates the median peak ground acceleration at which these ceramic insulators are lost to be approximately 0.25 g. NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty –Nine Nuclear Power Plants East of the Rocky

Mountains," provides an estimate for annual probability of exceedance for peak ground acceleration of approximately 2E-5 for GGNS and a ground acceleration of 0.25 g. Therefore, the seismic LOOP initiator frequency is approximately two orders of magnitude lower than the GGNS LOOP initiating event frequency (3.9E-2) times the GGNS four hour non-recovery probability of offsite power (6E-2) used in the GGNS base internal events PSA model (that is, 3.9E-2/yr X 6E-2 = 2.3E-3/yr). Based on this estimate and the relatively insignificant risk impact of the proposed changes to the internal events PSA model, the impact of the proposed changes to seismic risk is considered to be insignificant.

While PSA techniques were used to develop core damage frequencies associated with internal fires, the results from the IPEEE are still screening analyses and therefore are not directly comparable to the CDF results from the internal events PSA. The CDF values generated for the IPEEE were intended to show that the CDF is low enough that a vulnerability does not exist. The fire PSA was not developed to the same level of detail as the internal events. Therefore, the fire CDF reported in the IPEEE, as a general rule, should not be combined with, or directly compared to the internal events analysis. A review of the Fire PSA scenarios indicates that approximately 14.6% of the fire CDF (1.3E-6/year) is associated with a fire induced LOOP event. This is compared to a 42.5% contribution (2.3E-6/year) from LOOP initiators for the base internal events PSA. These frequencies are relatively close and since additional DG out of service time would primarily impact LOOP scenarios, the effect of the proposed change on fire CDF would be expected to be similar to the impact on the internal events PSA CDF. Since the impact of the change on internal events CDF and ICCDP is well under the acceptance guidance, there is no need to quantitatively evaluate the impact on fire risk. It is expected to be non-risk significant also.