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SUBJECT: Responds to Item 4 of 981117 RAI re TS change request 376 re extended EDG allowed outage time,in manner consistent with RGs 1.174 & 1.177.

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April 23, 1999

U.S. Nuclear Regulatory Commission
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Gentlemen:

In the Matter of	)	Docket Nos. 50-260
Tennessee Valley Authority	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 2 AND 3 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING TECHNICAL SPECIFICATIONS CHANGE NO. 376 - EXTENDED EMERGENCY DIESEL GENERATOR ALLOWED OUTAGE TIME (TAC NOS. M98205 AND M98206)

This is the response to Item 4 of the November 17, 1998, RAI regarding Technical Specifications (TS) change request 376. Responses to Items 1, 2, and 3 from the RAI were previously provided in a submittal dated March 30, 1999. TS-376 was originally submitted on March 12, 1997, and proposed changes to the Units 2 and 3 TS to extend the current 7-day Allowed Outage Time (AOT) for the Emergency Diesel Generators (EDGs) to 14 days.

Item 4 of the RAI asked that TVA provide certain additional information in a manner consistent with the guidelines of Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, July 1998", and RG 1.177, "An Approach for Using Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, August 1998". These Regulatory Guides were issued subsequent to the TS-376 submittal. The response to Item 4 is provided in Enclosure 1.

D030

The change in core damage frequency (CDF) provided in the March 12, 1997, TS-376 submittal was based on a very conservative contribution of EDG unavailability associated with the requested EDG extended AOT. In this submittal, the Probabilistic Safety Analysis (PSA) for the EDG AOT extension has been re-evaluated to incorporate the relevant guidance from RGs 1.174 and 1.177. This includes both revised CDF calculations and the results of Large Early Release Frequency calculations. Additionally, a discussion of PSA quality is included in Enclosure 1.

One commitment is provided in Enclosure 2. If you have any questions, please contact me at (256) 729-2636.

Sincerely,

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE - 376
EXTENDED ALLOWED OUTAGE TIME (AOT)
EMERGENCY DIESEL GENERATORS (EDGs)

RESPONSE TO ITEM 4 - NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)
DATED NOVEMBER 17, 1998

Below is the response to Item 4 of the subject RAI on TS-376. Responses to the other three NRC RAI items were provided in a submittal dated March 30, 1999. TS-376 was submitted on March 12, 1997, and proposed changes to the Units 2 and 3 TS to extend the current 7-day AOT for the EDGs to 14 days.

RAI Item 4 is repeated below *(shown in italics)* along with the TVA response for each sub-item.

NRC ITEM 4

The PSA Analysis section of your application includes information that is relevant to assessing the risk associated with the proposed change (e.g., baseline core damage frequency (CDF) and new CDF). However, the subject section lacks other pertinent information (e.g., probability risk assessment (PRA) quality and three-tiered approach[1] for licensees to evaluate the risk associated with proposed technical specification (TS) allowed outage time changes).

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[1] *Tier 1 is an evaluation of the impact on plant risk of the proposed technical specification change as expressed by the change in core damage frequency, the incremental conditional core damage probability, the change in large early release frequency, and the incremental conditional large early release probability. Tier 2 is an identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk significant operational factors such as concurrent system or equipment testing were also involved. Tier 3 is the establishment of an overall configuration risk management program to ensure that other potentially lower probability, but nonetheless risk-significant, configurations resulting from maintenance and other operational activities are identified and compensated for. A detailed discussion of the three-tiered approach is provided in Regulatory Guide (RG) 1.177.*

The staff has recently published several RGs which discuss approaches for risk-informed decision-making and staff's expectations for using PRA in licensing actions. For the proposed change, TVA should provide the staff with information that is consistent with the guideline provided in RG 1.177, An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications, and RG 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis. TVA should refer to RG 1.177 for a comprehensive and detailed guideline to provide sufficient information required for a staff review. As described in RG 1.177, TVA should address the questions posed for the three tiers as well as questions addressing the quality of PRA. In adhering to RG 1.177, TVA should develop a configuration risk management program and incorporate it in the proposed risk-informed TSs.

<u>TVA Response to Item 4 - Additional Information on Probabilistic Safety Assessment (PSA) Results including Revised Tier 1 Results and PSA Quality</u>

Subsequent to the March 12, 1997, submittal of TS-376, Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", and 1.177, "An Approach for Using Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" were issued. These RGs provide NRC guidance regarding the use of PSA to support TS changes for extended AOTs and extended surveillance test intervals.

The change in core damage frequency (CDF) provided in the March 12, 1997, TS-376 submittal was based on a very conservative contribution of EDG unavailability resulting from the requested EDG extended AOT. In response to NRC's request in Item 4, the PSA analysis for the EDG AOT extension has been re-evaluated to incorporate the relevant guidance from RGs 1.174 and 1.177. This includes both revised CDF calculations and Large Early Release Frequency (LERF) calculations. A discussion of PSA quality is also included in this section.

<u>Current PSA Models</u>

BFN currently utilizes two PSA models (RISKMAN software) for the evaluation of the risk significance of proposed plant changes. The PSA models are based on the Unit 2 PSA originally submitted to NRC on September 1, 1992, in response to Generic Letter 88-12, "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities, November 1988", and the associated IPE activities.

The current PSA models are:

- The Unit 2 PSA Model with Unit 3 Operating, and
- The Unit 3 PSA Model with Unit 2 Operating

These two PSA models were provided to NRC in support of the NRC Maintenance Rule inspection. The PSA models have been reviewed by the PSA Peer Review Certification team, sponsored by the Boiling Water Reactor Owner's Group (BWROG), as described in more detail in the PSA Quality section.

Calculation of EDG Unavailability Due To 14-Day EDG AOT

The 14-day EDG total unavailability associated with the 14-day AOT is the sum of the following terms:

- Actual (pre-extended AOT) EDG unavailability
- Increase in unplanned EDG unavailability
- Increase in planned EDG unavailability

Actual BFN EDG unavailability statistics have been updated from the Maintenance Rule database from June 1994 through February 1999 for all eight EDGs. This data also includes the actual unavailability due to the 12-year vendor recommended preventive maintenance (PM) outages for all 8 EDGs since these 12-year outages occurred during this period.

For this time period, the data shows the total EDG actual unavailability was 0.01660 per EDG comprised of an average planned unavailability of 0.01374 (83%) and an average unplanned unavailability of 0.00286 (17%). The 12-year vendor recommended PM contribution to the net unavailability was an average of 120 hours per EDG. Since the data does not include a full 12 years of data (the data includes an average of over 4 years), the actual unavailability value is higher than normal since the 12-year vendor outages are included. The 12-year PM is the longest of all planned PM outages.

The projected increase in EDG unavailability resulting from the 14-day AOT extension was calculated using the following guidance from RG 1.177:

> The average downtime can be assumed to proportionally increase with the increase in the proposed AOT for downtimes associated with unscheduled maintenance. For scheduled

(preventative) maintenance, the downtime assumed can be representative of plant practices (e.g., one-half of the AOT):

Therefore, the increase in unplanned EDG unavailability is:

= [(extended AOT - original AOT)/(original AOT)] x (normal unplanned unavailability)

= [(14-7)/7] x (0.0029) = 0.0029

Regarding plant practices for planned unavailability associated with the 14-day DG AOT, the 12-year vendor recommended PM is the only PM that potentially has an effect since the remaining PMs can easily be completed within the current 7-day AOT. In the calculation of the increase in planned unavailability, it is, however, appropriate to subtract the 12-year maintenance average duration, which averaged 120.0 hours (5 days) per EDG, from the unplanned unavailability since it is already accounted for in the actual EDG unavailability term.

The increase in planned unavailability is thus calculated as follows:

= (Maximum # days per AOT - 5) / [(# days per year) x (# of years per performance)]

= (14 - 5)/ (365 x 12) = 0.00205

To generate the total unavailability due to the 14-day EDG AOT the three unavailability terms are summed:

= Normal EDG unavailability + Increase in unplanned EDG unavailability + Increase in planned EDG unavailability

= 0.0166 + 0.0029 + 0.00205

= 0.02155

Tier 1 Calculation Results

As described in RG 1.177, Tier 1 is an evaluation of the impact on plant risk of the proposed TS change as expressed by the change in CDF, the incremental conditional core damage probability (ICCDP), and when appropriate, the change in the LERF and the Incremental Conditional Large Early Release Probability (ICLERP). The EDG unavailability (UA) data was used to quantify the Unit 2 and Unit 3 models using RISKMAN software for both the normal EDG unavailability case and the 14-day AOT extension case for CDF and LERF.

The change in CDF and LERF due to the 14-day AOT are in the Table below for the Unit 2 and Unit 3 PSA models.

CDF/LERF RESULTS				
CASE	UNIT 2 CDF	UNIT 2 LERF	UNIT 3 CDF	UNIT 3 LERF
Normal EDG (UA = 0.0166)	4.6952E-6	1.5778E-6	7.8681E-6	2.5163E-6
14-Day AOT (UA = 0.02155)	4.7404E-6	1.5975E-6	7.9592E-6	2.5365E-6
Change (Δ)	4.52E-8	1.97E-8	9.11E-8	2.02E-8
Change (%)	0.96%	1.25%	1.16%	0.80%

ICCDP is defined by RG 1.177 as:

ICCDP = [(Conditional CDF w/ equipment out of service) - (Baseline CDF w/ nominal expected equipment unavailabilities)] x (duration of single AOT under consideration)

The conditional CDF is calculated by multiplying the baseline CDF by the Risk Achievement Worth (RAW) for the EDG out of service. The RAW used is the average RAW for the base unavailability case (UA = 0.0166). The value for RAW is 1.41 for both Unit 2 and 3. The duration of the AOT under consideration is 7 days, which is the difference between the current AOT and the requested AOT (this must be adjusted for the duration on a yearly basis by dividing by 365).

Therefore, ICCDP is derived as:

ICCDP = [(RAW x (Baseline CDF)) - (Baseline CDF)] x (duration of single AOT under consideration)

Since the ratio of LERF/CDF is relatively constant especially for such a small change in CDF, the ICLERP can be estimated by dividing LERF by CDF for each units base case and multiplying the result by the ICCDP which shown as an equation is:

ICLERP = (LERF/CDF) x ICCDP

The results of the Δ CDF/ICCPD and Δ LERF/ICLERP risk measures calculations are presented in these tables:

CDF/ICCPD Results		
CASE	UNIT 2	UNIT 3
Base Case (UA = 0.0166)	4.6952E-6	7.8681E-6
14-Day AOT Case (UA = 0.02155)	4.7404E-6	7.9592E-6
Δ CDF	4.52E-8	9.11E-8
NRC GUIDANCE Δ CDF	< 1E-6	< 1E-6
ICCDP	3.69E-8	6.19E-8
NRC GUIDANCE ICCDP	< 5.0E-7	< 5.0E-7
% CDF Increase	0.96%	1.16%

LERF/ICLERP Results		
CASE	UNIT 2	UNIT 3
Base Case (UA=0.0166)	1.5778E-6	2.5163E-6
14-Day AOT Case (UA=0.02155)	1.5975E-6	2.5365E-6
Δ LERF	1.97E-8	2.02E-8
NRC GUIDANCE Δ LERF	< 1E-7	< 1E-7
ICLERP	1.24E-8	1.98E-8
NRC GUIDANCE ICLERP	< 5.0E-8	< 5.0E-8
% LERF Increase	1.25%	0.80%

Conclusions

As can be seen from the above tables, the changes due to extension of the current 7-day EDG AOT to 14 days are risk insignificant and well below NRC acceptance criteria specified in RG 1.174 and 1.177.

PSA QUALITY

During November 1997, TVA participated in a PSA Peer Review Certification of the BFN PSA administered under the auspices of the BWROG Peer Certification Committee. The purpose of the PSA Peer Review process is to establish a method of assessing the technical quality of the PSA for the spectrum of its potential applications.

The BFN PSA Peer Review Certification team consisted of six individuals with a combined 134 man-years of nuclear experience including 97 man-years in PSA related applications. These engineers and analysts provided both an objective review of the PSA technical elements and a subjective assessment based on their PSA experience. The review team had considerable expertise in basic PSA development and PSA applications, and in the specific PSA methodology used for the BFN PSA. The team was also knowledgeable in BWR-4 plant design and operational practices.

The evaluation process used a tiered approach of standard checklists that allowed for a detailed review of the elements and the sub-elements of the BFN PSA to identify strengths and areas that needed improvement. A review system was used that allowed the Peer Review team to focus on technical issues and to rate their assessment results in the form of a "grade" of 1 through 4 on a PSA sub-element level. To reasonably span the spectrum of potential PSA applications, the four grades of certification as defined by the BWROG document, "Report to the Industry on PSA Peer Review Certification Process: Pilot Plant Results," were employed. These are repeated below for reference.

GRADING SYSTEM

Grade 1 - Useful for Identifying Severe Accident Vulnerabilities, Accident Management Insights, and General Prioritization of Issues

This grade represents the minimum standard and has satisfied NRC expectations for responding to Generic Letter 88-20. Most PSAs are expected to be capable of meeting these requirements. This grade of certification would serve as an industry standard.

Grade 2 - Useful for Risk Ranking Supplemented by Deterministic Input

This grade of certification requires a review of the PSA model, documentation, and a maintenance program. Certification at this grade would provide assurance that, on a relative basis, the PSA methods and models yield meaningful rankings for the assessment

of systems, structures, and components, when combined with deterministic insights (i.e., a blended approach).

Grade 3 - Useful for Risk Significance with Deterministic Input

This grade of certification extends the requirements to assure that risk significant determinations made by PSA using absolute risk insights are adequate to support a broader range of regulatory applications when combined with deterministic insights.

Grade 4 - Useful as a Primary Basis for Decision Making

This grade of certification requires a comprehensive, intensively reviewed study, which has the scope, level of detail, and documentation to assure the highest quality of results. Routine reliance on the PSA as the basis for certain changes is expected as a result of this grade. It is expected that few plants would currently be eligible for this grade of certification.

It should be noted that while each of the four application oriented grades have different characteristics as previously delineated, the boundaries between the grades are not sharp. This leaves, in some cases, an element of judgment to be applied when assigning a specific application to a specific grade. This lack of sharp boundaries is due in part to the fact that varying degrees of supplementary deterministic considerations or focused PSA studies may be used with any of the four grades of PSA to effectively support an application.

Results of BFN Peer Review

The BFN PSA Peer Review provided a consistent evaluation across all the PSA elements and sub-elements. Approximately 72% of the graded sub-elements were at Grade 3 or above; 8% of the sub-elements were assessed at Grade 4 providing a very solid evaluation. The following Table summarizes the results of the BFN Peer Review performed at the element level for the BFN PSA.

RESULTS OF BEN PEER REVIEW		
PSA ELEMENT	CERTIFICATION GRADE	COMMENTS ON GRADE 2 ELEMENTS
INITIATING EVENTS (IE)	3	
ACCIDENT SEQUENCE EVALUATION (AS)	3	
THERMAL HYDRAULIC ANALYSIS (TH)	2	Further documentation would fully support Grade 3 classification
SYSTEMS ANALYSIS (SY)	3	
DATA ANALYSIS (DA)	2	Use of plant specific data needed for Grade 3 classification
HUMAN RELIABILITY ANALYSIS (HR)	3	
DEPENDENCY ANALYSIS (DE)	3	
STRUCTURAL RESPONSE (ST)	3	
QUANTIFICATION (QU)	3	
CONTAINMENT PERFORMANCE ANALYSIS (L2)	2	Would support Grade 3 classification with several enhancements
MAINTENANCE AND UPDATE PROCESS (MU)	3	

Since the 14-day EDG AOT application is fully supported by elements associated with Grade 3, only the elements associated with Grade 2 are discussed below with regard to the EDG AOT extension request.

Thermal Hydraulic Analysis (TH)

Peer Team TH Observation: The thermal hydraulic inputs to the model are generally based on best estimate calculations although the calculations were not always referenced or readily available for review. These calculations were diverse in nature: hand calculations, Modular Accident Analysis Program (MAAP) runs, Updated Final Safety Analysis Report calculations, and NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants Final Summary Report" analyses. The key recommendation is to maintain documentation of supporting MAAP calculations. Further documentation is required for this element to be supportive of Grade 3 applications.

TH Evaluation by BFN: Since the concern is documentation based and not technical adequacy, the TH element is fully supportive of the EDG AOT extension evaluation. Improvements in documentation to fully support a Grade 3 element are planned to be incorporated in the next PSA update currently scheduled to be completed in March 2000.

Data Analysis (DA)

Peer Team Observation: Guidance for data analysis is contained in the IPE submittal text and is sufficient to allow updating. The majority of the PSA data is, however, generic. Plant-specific data analysis is limited primarily to EDG failure rates and surveillance unavailabilities. The common cause failure analysis is performed per NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies Analytical Background and Techniques", and is generally thorough and rigorous. The key recommended enhancement is the use of plant specific data in the component failure rates and unavailabilities to replace the existing generic data. This enhancement would be required to fully support Grade 3 applications.

DA Evaluation by BFN: Plant specific data relating to EDG unavailability was used in the EDG AOT evaluation. Therefore, this element is fully supportive of the EDG AOT extension evaluation. The replacement of generic data with plant specific data will be incorporated in the next PSA update which is currently scheduled to be completed in March 2000.

Containment Performance Analysis (L2)

Peer Team L2 Observations: The Level 2 model addresses most of the important phenomena. The Containment Event Tree is logical. There is a manual interface required to ensure Level 1 to 2 interface occurs. There is an excellent use of MAAP to provide severe accident insights. The Level 1/2 interface has a defined process to transfer information; however, some simplification in modeling occurs at this point, leading to some conservative and potentially some non-conservative treatments. There is ambiguity regarding the definition of the point where the transition between Level 1 and 2 occurs, i.e., definition of core damage appears different in the two segments of the study. The key phenomena associated with severe accidents are included in the Level 2 analysis. There is a need to increase the visibility of certain phenomena, system interactions, and Human Error Probability treatment.

The Level 2 end states are adequate for understanding the relationship to source term impacts on the public health and

safety, and the Level 2 end states are technically adequate to represent these impacts.

LERF needs to be defined based on Level 2 end states. The LERF definition may require assurance that the release category definition of Early is consistent with the Emergency Plan Emergency Action Levels. The Level 2 analysis is fully supportive of Grade 2 applications, but several key enhancements to the analysis are necessary for this element to be supportive of Grade 3 applications.

L2 Evaluation by BFN: As indicated in the above observation, the Level 2 end states were considered technically adequate, although the observation indicated LERF needs to be defined based on Level 2 end states. The PSA models were subsequently modified to define LERF based on Level 2 end states. Therefore, this part of the observation recommendation has been implemented and was used for the EDG AOT LERF calculation in this response.

The remaining L2 observations do not affect the EDG AOT evaluation. These concerns will be remedied in the next PSA update currently scheduled to be completed in March 2000.

Conclusions

The Peer Review team observations were used to enhance the PSA evaluation of the subject EDG AOT extension request in that it provided a focus on issues that had potential to impact the calculation of risk. In summary, TVA concludes that the BFN PSA model used for evaluating the risk change in the EDG AOT extension request is appropriate and adequate to support the 14-day EDG AOT TS amendment request.

NRC Item 4.a

In addressing Tier 2 and Tier 3 requirements, also include discussions on the following specific issues:

a. *Because of the potential safety impact of the extended EDG allowed out-of-service time (AOT) for PM, the staff believes that certain compensatory measures are needed during the extended EDG AOT to assure safe operation of the plant. Provide a discussion of how each condition listed below would be addressed:*

   1. *The TSs should include verification that the required systems, subsystems, trains, components, and devices that depend on the remaining EDG as a source of emergency power are operable before removing an EDG for PM. In addition, positive measures should be provided to preclude testing*

or maintenance activities on these systems, subsystems, trains, components, and devices while the EDG is inoperable.

2. *Removal from service of safety systems and important non-safety equipment, including offsite power sources, should be minimized during the outage of the EDG for PM.*

3. *Component testing or maintenance that increases the likelihood of a plant transient should be avoided; plant operation should be stable during the EDG PM.*

TVA Response to Item 4.a

Tier 2

As described in RG 1.177, Tier 2 is an identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk significant operational factors such as concurrent system or equipment testing were also involved.

BFN has an active and comprehensive risk management program which is described in the following plant Standard Programs and Processes (SPPs) and Technical Instructions (TI):

- SPP-7.0, Work Control and Outage Management
- SPP-7.1, Work Control Process
- SPP-7.2, Outage Management
- 0-TI-367, BFN Dual Unit Maintenance

For on-line maintenance, risk is controlled through a 12-week rolling schedule. A schedule of sequenced work windows is established for on-line periods when combinations of plant systems can acceptably be out of service to perform PM and surveillance activities. The predetermined work windows incorporate risk assessments to determine potential impacts to the safe and reliable operation of the unit and assures long-term maintenance activities are performed within required frequencies to maximize plant equipment and component availability. Within each schedule week, proposed system configurations are compared against the dual unit maintenance risk matrix contained in 0-TI-367 to determine if the proposed configuration is risk significant.

The dual unit maintenance risk matrix is based on the current BFN PSA models and is color coded to alert the work scheduler to potential risk significant plant configurations. Prior to scheduling any work which could result in a risk significant configuration per the dual unit maintenance matrix, Engineering is contacted to perform a more detailed evaluation of the proposed configuration for risk significance. Similarly, if emergent work results in a risk significant configuration, a priority work request is generated to correct the situation and Engineering would evaluate the situation for risk significance. Additionally, BFN has also recently installed SENTINEL, an Electric Power Research Institute (EPRI) on-line risk management computer program tool to assist in the management of on-line risk.

Outage risk management is controlled with the use of outage risk assessment reviews which utilizes ORAM software, an EPRI outage risk assessment computer program. ORAM is further augmented by an Independent Safety Review Team, which is composed of at least six members with broad and detailed plant knowledge and knowledge of shutdown safety issues affecting the nuclear industry. This team reviews the unit Outage Plan and the detailed schedule to ensure that all shutdown safety issues are addressed and all reasonable actions have been taken to minimize shutdown risk.

Based on BFN's current risk management program described above, and in consideration of the insignificant change in CDF and LERF due to the proposed AOT (approximately 1%), we consider current BFN practices and procedures adequately address Tier 2 concerns.

Tier 3

TVA has several performance goals which emphasize minimizing EDG unavailability and maximizing reliability. The current Maintenance Rule EDG unavailability performance criteria is .0342. An EDG unavailability goal of .015 has been established as an Institute of Nuclear Power Operations performance goal for 1999. The new NRC performance indicator for EDGs in SECY-99-007, "Recommendations for Reactor Oversight Process Improvements, January 1999", also uses .015 as the EDG unavailability threshold for increased regulatory response. The EDG unavailability for FY 1999 through February is .005.

From these goals, it is clear that there is considerable emphasis on maximizing EDG availability which can only be achieved by minimizing EDG outage time and maintaining the EDGs in good working condition. Therefore, while the proposed TS relaxes the current 7-day AOT, actual EDG unavailability is not apt to rise

markedly due to the emphasis placed on meeting the various EDG performance goals. Hence, use of the AOT extension is likely only in rare cases.

RG 1.177 recommends that a formal Tier 3 Configuration Risk Management Program (CRMP) be developed and implemented for systems for which a PSA AOT extension has been granted to identify possible risk significant configurations under Tier 2 that could be encountered over extended periods of time. However, as noted above, TVA does not expect to use the extended EDG AOT except for infrequent unexpected major corrective EDG maintenance and as a contingency provision for PM activities. Of the PM activities, only the 12-year vendor recommended PM has an expected potential need for an extended AOT should the maintenance encounter unexpected complications.

Therefore, the scope and applicability of a CRMP would be extremely narrow at this time since it would apply only to the EDG system when using the extension. Considering the narrow scope and the low risk factors associated with the extended EDG AOT as evidenced by the small increase in CDF and LERF, we do not believe that development of a RG 1.177 type CRMP is necessary at present based on the EDG AOT extension alone.

Since a CRMP will likely be required in conjunction with the proposed change to Section a(3) of 10 CFR 50.65 (Maintenance Rule), which would have a much broader applicability, we believe it is appropriate to postpone development of a CRMP pending implementation of the revised Maintenance Rule. We note also that NRC guidance on risk management for maintenance activities is still in draft format (Draft Regulatory Guide DG-1082). Postponement would allow a more orderly development of a CRMP consistent with NRC standards and nuclear industry application and may eliminate the need for a future license amendment to revise the CRMP based on the change to the Maintenance Rule.

NRC Item 4.b

b. *The condition of off-site sources of electrical power prior to and during the extended EDG outage time have additional importance. Discuss what considerations should be given to not performing the extended maintenance when the offsite grid condition or configuration is degraded or when adverse or extreme weather conditions (e.g., high winds, lightning, icing) are expected. Discuss how planning of the extended EDG maintenance should consider the time needed to complete the maintenance and the ability to accurately forecast weather conditions that are expected to occur during the maintenance. Discuss what, if any, contingency plans should be developed to restore the inoperable EDG in the event of*

unanticipated adverse weather or degraded grid conditions occurring which can significantly increase the probability of losing offsite electrical power.

TVA Response to Item 4.b

As discussed in the March 12, 1997 submittal of TS-376, based on the Nuclear Utility Management and Resource Council (NUMARC) 87-000, "Guidelines for and Technical Basis for NUMARC Initiatives for Addressing Station Blackout at Light Water Reactors", criteria for characterizing the susceptibility of plants to Loss of Off-site Power (LOOP) events for the Station Blackout rule, BFN is in the category of plants least likely to lose off-site power because of extremely severe weather. NUMARC 87-000 criteria also defines BFN as a Group I ½ site which is the least susceptible category to LOOP events due to grid-related disturbances. This favorable categorization is based on physical separation of BFN switchyards and off-site transmission lines. Therefore, BFN is unlikely to experience a LOOP event due to weather or grid related phenomena.

BFN safety philosophy would not allow risk significant maintenance to be planned or performed during periods when extreme bad weather is expected. In response to NRC's concern in Item 4.b, TVA will formalize this safety philosophy by revising plant procedure, 0-TI-367, BFN Dual Unit Maintenance, to prescribe that a scheduled EDG maintenance outage (outage length greater than 7 days) will be rescheduled if severe weather conditions are predicted. Requirements for restoring an inoperable EDG to service in the event of unexpected severe weather or degraded grid conditions are dependent on the condition of the EDG at the time. Shift Operations management has the procedural authority to direct site resources as deemed necessary to restore the equipment to an operable status as soon as possible if weather or other circumstances so dictate.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE - 376
EXTENDED ALLOWED OUTAGE TIME (AOT)
EMERGENCY DIESEL GENERATORS (EDGs)

RESPONSE TO ITEM 4 - NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)
DATED NOVEMBER 17, 1998

COMMITMENT LIST

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1. TVA will revise plant procedure, 0-TI-367, BFN Dual Unit Maintenance, to prescribe that a scheduled EDG maintenance outage (outage length greater than 7 days) will be rescheduled if severe weather conditions are predicted. (This commitment contingent on NRC approval of TS-376).