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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

July 12, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. E. J. Butcher, Jr., Chief
Operating Reactors Branch #3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Calvert Cliffs Nuclear Power Plant
Units Nos. 1 & 2; Dockets Nos. 50-317 and 50-318
Request for Additional Information Following Preliminary
Staff Review of BG&E Responses to Generic Letter 83-28

References: (a) Letter from D. G. Eisenhut, to All Licensees, dated July 8, 1983, Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events"

(b) Letter from A. E. Lundvall, Jr., to D. G. Eisenhut, dated February 29, 1985, same subject

(c) Letter from J. R. Miller, to A. E. Lundvall, Jr., dated April 2, 1985, same subject

(d) Letter from A. E. Lundvall, Jr. to J. R. Miller, dated June 7, 1985, Request for Additional Information.

Gentlemen:

This letter completes our response to Reference (c). A partial response was provided in Reference (d).

In Reference (b), we committed to the CE Owners Group program to perform an overall reliability assessment of the CE reactor trip system (RTS) design in order to identify any components which are reliability-sensitive to failure rate uncertainty and testing considerations as outlined in Reference (a).

The CE Owners Group program was completed in December 1984, and based on our review of the results of that program, we have concluded that the current Calvert Cliffs RTS test intervals are consistent with achieving high RTS availability.

The methodology used in the RTS reliability assessment program was as follows. A fault tree model was constructed for a failure to trip the reactor on

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demand. The model explicitly addressed random component failures, common cause failures, operator errors and out-of-service time for testing. Component failure rates were quantified using applicable operating experience data (LERs, NPRDS) to perform a Bayesian update of WASH-1400 failure rate distributions. Common cause failure rates were quantified using operating experience data and the Vesely specialization of the Marshall-Olken algorithm. The fault tree model was quantitatively evaluated using Monte Carlo simulation to derive a system unavailability distribution. A sensitivity analysis was performed to determine the sensitivity of overall system unavailability to variations in the failure rates of individual components. Testing frequencies and down-times were derived from the technical specifications. Functional demands were estimated based on scram/shutdown data compiled in CE's Reliability Data System.

The results of this analysis showed that the median probability that the RTS will fail to trip the reactor is less than 4×10^{-6} per demand, and the 95th percentile confidence limit probability is less than 9×10^{-6} per demand for the Calvert Cliffs class of CE plant. Based on these results and the NRC's estimate of 2×10^{-5} as the probability that the RTS would fail to trip the reactor, we have concluded that the current design and test interval provide a high degree of RTS reliability.

If you have any questions regarding this response, please do not hesitate to contact us.

Very truly yours,

A. E. Lundvall Jr.

AEL/RCLO/vf

cc: D. A. Brune, Esq.
G. F. Trowbridge, Esq.
Mr. D. H. Jaffe, NRC
Mr. T. Foley, NRC

FORM: 97888110
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ENGINEERING REVIEW AND SAFETY ANALYSIS
NUCLEAR POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

DESIGN CHANGE TITLE / STATION / UNIT: (1 DRY CASK INDEPENDENT SPENT FUEL STORAGE INSTL./SURRY UNITS 1&2	DESIGN CHANGE NO.: (2 84-53
PREPARING ENGINEER/AFFILIATION: (3 P.V. Holton /BECHTEL ASSOCIATES PROFESSIONAL CORP. (VA)	DATE: (4 12/10/84
REVIEWING ENGINEER/AFFILIATION: W.A. 12/13/84 (5 /BECHTEL ASSOCIATES PROFESSIONAL CORP. (VA)	DATE: (6 12/10/84
LEAD ENGINEER/AFFILIATION: (7 /Vepco	DATE: (8 12/12/84
DESIGN CONTROL ENGINEER: (9	DATE: (10 12/17/84
STATION NUCLEAR SAFETY AND OPERATING COMMITTEE APPROVAL: H L Miller (11	DATE: (12 DEC 20 1984
SEC INDEPENDENT SAFETY REVIEW: (13	DATE: (14

ENGINEERING REVIEW AND SAFETY ANALYSIS (SHALL CONSIST OF): (15
1. STATEMENT OF PROBLEM 2. IDENTIFICATION OF QUALITY GROUP AND CATEGORY 3. PROPOSED RESOLUTION (INCLUDES ANALYSIS OF FIRE HAZARDS, SEISMIC AND ENVIRONMENTAL QUALIFICATION, ALARA CONSIDERATIONS, RECENT NRC CONCERNS, AND IMPACT OF/ON OTHER DESIGN CHANGES) 4. REVIEW OF TECHNICAL SPECIFICATIONS 5. REVIEW OF UPDATED FINAL SAFETY ANALYSIS REPORT 6. DESIGN BASIS DOCUMENT REVIEW 7. UNREVIEWED SAFETY QUESTIONS 8. SAFETY AND OPERATIONAL IMPLICATION

1.0 STATEMENT OF PROBLEM

This modification is necessary to initiate the construction of the Independent Spent Fuel Storage Installation (ISFSI) in the Surry Power Station (SPS). The design criteria for the Sealed Surface Storage Cask System (SSSCs) will be applied to the Surry ISFSI. These criteria will be in compliance with the requirements set forth in 10 CFR Part 72, as demonstrated in the Surry ISFSI Safety Analysis Report.

This facility, when completed, will provide additional interim storage capacity for the spent fuel resulting from the operation of the two pressurized water reactors at the Surry Power Station. Physical characteristics of the fuel to be stored in the ISFSI are summarized in Chapter 3 of the Surry ISFSI Safety Analysis Report (SAR).

This package will include only the construction of the non-safeguard portion of the ISFSI slabs, roads, perimeter fence, emergency diesel, lighting and associated ductbanks. The facility will have a total of three slabs when finished. This Design Change Package (DCP) will provide for construction of the first of the three. Later work will be initiated by future DCPs. Furthermore, the design, fabrication, handling operations, and maintenance of the storage casks are outside the scope of this DCP.

2.0 IDENTIFICATION OF QUALITY GROUP AND CATEGORY

In accordance with the VEPCO ISFSI SAR (Section 3.4) and per the definitions in the Nuclear Power Station Quality Assurance Manual (NPSQAM), Appendix A, the structures and components installed by this design change are non-seismic and are not safety-related. Specifically, the electrical components have non-IE status. Consequently, the structures and components addressed in this DCP are designated NPSQAM Category III.

ENCLOSURE 5

3.0 PROPOSED RESOLUTION

As stated in Section 1.0, Statement of Problem, the work involved in this modification is construction of roads, storage pads, fences, emergency diesel, lighting, and ductbanks. No special problem resolution is required.

Fire Hazard Analysis

The construction work in this modification is away from the Surry Power Station (SPS), thus away from any fire area associated with the station. Although combustible material such as diesel oil will be added resulting from this modification, this package will not increase the probability or consequence of a fire in SPS since the distance from the SPS is approximately 3000 feet.

The diesel generator fuel tank holds approximately 185 gallons of No. 2 diesel fuel. To prevent the contamination of the nearby game preserve, a curb has been designed. The curb will contain any fire to the area in the vicinity of the diesel generator. The tank is provided with a flame arrestor and there are no normal ignition sources in the immediate area. There are no governing requirements in the NFPA to mandate fire protection at the diesel generator fuel tank. A portable wheeled dry chemical fire extinguisher of 200 to 300 lb. capacity with hose and shutoff nozzle is recommended to be kept nearby.

Seismic Analysis

All components associated with this modification are non-seismic and non-safety related, therefore, no safety related structures or equipment would be affected during a seismic event as a result of this modification. No margin of safety for structures, systems, and components will be decreased by this modification in the event of a seismic event.

Environmental Qualification

Since no component in this design package is Class 1E, environmental qualification requirements for Class 1E equipment has not been required. However, the equipment is qualified for its intended function and environment.

ALARA Analysis

The construction work prescribed by this DCP will be performed in an area near the Low Level Waste Storage Facility (LLWSF). As such, this area has the potential for being a radiation area; however, the station has indicated that at the time of construction, this area will be a non-radiation area. Therefore, an ALARA analysis concerning construction related activities need not be performed for this section.

A comprehensive ALARA analysis for the operational phase of the ISFSI has been prepared as part of the Surry ISFSI SAR and ER, and in response to NRC questions. This ALARA analysis (pending formal NRC approval of the

ISFSI) has been sufficient to show that operation of the ISFSI is ALARA. Therefore, further analyses are not needed for this DCP. However, the design and operational considerations which were included in the SAR's ALARA analysis, as well as its preliminary dose estimates, are summarized here for completeness.

Maintenance

A minimum amount of equipment is being placed in the vicinity of the ISFSI. That equipment which is being provided adjacent to the ISFSI is being specified with high-reliability in mind to minimize maintenance. Equipment is being located as far away from the ISFSI as practical within the constraints of security and operations.

Corrosion Products

The formation of activated corrosion products is precluded since there are no releases associated with the operation of the ISFSI.

Shielding

Shielding is provided by the massive casks to be placed at the ISFSI. Other permanent shielding is not considered practical or necessary. Should long duration maintenance be necessary at the diesel generator, sufficient space is available to erect temporary shielding such as sand bags.

Dose Estimate

The annual doses due to ISFSI operations have been estimated in the Surry ISFSI SAR and ER as follows:

Annual Doses from ISFSI Operations

	Man-Rem
LLWSF*	9.8
Surry Power Station*	1.5
ISFSI Operations -	
Cask Preparation and Placement**	13.0
Maintenance and surveillance	0.3
Total	24.6

*Assumes completed ISFSI (84 casks)
**Assumes 4 casks per year

See Chapter 7 of the Surry ISFSI SAR for more details on the overall ALARA aspects of operations at the ISFSI.

ENGINEERING REVIEW (CONTINUED): (15

An Engineering ALARA Design Guide has been filled out in accordance with STD-NEN-0001 and is attached to this DCP.

Recent NRC Concerns

Several recent meetings with responsible NRC staff representatives indicate no concerns applicable to this design change package.

Impact on Other Design Changes

The modifications in this design change are interrelated with DCP 84-52, and have been coordinated appropriately. Further discussion of this interrelationship is not provided here because DCP 84-52 is a safeguards (security) DCP.

4.0 REVIEW OF TECHNICAL SPECIFICATION

Using the Table of Contents for the Technical Specifications, no technical specification requirements are imposed on any modification in this design change package, and no technical specification changes are necessary.

5.0 REVIEW OF UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

Using the keyword Index in the Surry UFSAR, modification in this DCP do not affect the description in the current UFSAR. No UFSAR change is required. However, for cross reference purposes, it is recommended that Figure 15.1-1, Site Plan be updated per the attachment.

6.0 DESIGN BASIS DOCUMENT REVIEW

Sections S1, S9, S12, E1, E3 and E8 of the Design Basis Document have been reviewed. The proposed modifications in this design change are either in compliance with the design basis or do not apply. No change to the Design Basis Document is required. However for cross reference purposes, it is recommended that Section S1 be updated per the attached.

7.0 UNREVIEWED SAFETY QUESTION EVALUATION

The modifications in this design change package do not constitute an "unreviewed safety question" as defined in 10CFR 50.59. These modifications do not:

a. Increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety as previously evaluated in the UFSAR since the modification will not affect the operation of any equipment or systems required to mitigate a Design Basis Accident.

ENGINEERING REVIEW (CONTINUED): (15

b. Create the possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR. This modification will neither replace nor modify any safety system, so the possibility of an accident of a different type than evaluated in the UFSAR is not created.

c. Reduce the margin of safety as defined in the basis for any Technical Specification. This modification does not alter the basis for any Technical Specification because no safety related component, system, structure or operation is affected.

8.0 SAFETY AND OPERATIONAL IMPLICATIONS

This design change is constructed away from SPS, and has no effect on the operation of any safety related equipment or structures. It has no impact on existing training, simulator, or maintenance requirements.

All aspects of this modification can be performed during any mode of reactor operations since no safety related component are being installed or are affected.