JUL 24 1981

SAFETY EVALUATION REPORT BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 2

TO CONSTRUCTION PERMITS CPPR-108 AND CPPR-109

Introduction

On December 19, 1980, Georgia Power Company, acting on its own behalf and agent for Oglethorpe Electric Membership Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, requested an amendment to the Construction Permits CPPR-108 and CPPR-109, for the Alvin W. Vogtle Nuclear Plant, Units 1 and 2, to reflect a modification in plant design. The modification would remove the enclosure building in the application for Vogtle licenses and add as a substitute a steel-framed, metal-siding equipment building from grade to the 270-foot level, and add a more restrictive primary containment leakage rate of 0.2 weight percent (%) per day. These substitutions, together with onsite meteorological data, will ensure that offsite post-accident doses are less than doses judged to be acceptable for design and are lower than those calculated at the time construction was initially authorized.

In support of this application for amendment, Georgia Power Company had previously submitted Supplement No. 6, dated August 21, 1979, to its application for Construction Permit and Operating License, which included a description of the associated physical modifications. These modifications would entail removal of the enclosure building and its related equipment as described in Sections 1.2.6, 3.8.4.1.1, and 6.6 of the Vogtle PSAR, while retaining from grade to the 270-foot level, an equipment building described in Sections 1.2.6 and 3.8.4.1.1 of Supplement No. 6.

To further support this request for amendment, Georgia Power Company has submitted Supplement No. 8 to the application, dated December 30, 1980, which contains additional information on the more restrictive primary containment leak rate and the resultant projected offsite doses for a design basis accident if the enclosure building and its related equipment were removed. Georgia Power Company has committed to a containment leak rate of 0.2% per day with the proposed modified enclosure building design. Employing this leakage rate, dispersion factors using NRC methodology and the latest onsite meterological data, Georgia Power Company has submitted revised design basis accident offsite doses for the Vogtle Units.

Based on the information submitted in the Preliminary Safety Analysis Report (PSAR) and Supplement Nos. 6 and 8 to the PSAR, the NRC staff has completed its review of all safety-significant matters related to the issuance of the construction permit amendment as requested in the December 19, 1980 application. This Safety Evaluation is issued in support of Amendment No. 2 to Construction Permits CPPR-108 and CPPR-109, allowing the removal of the enclosure building and related equipment.

The purpose of this Safety Evaluation is to examine the impact of the proposed modifications to the enclosure building for Vogtle Units 1 and 2. Specifically, the Safety Evaluation addresses the following safety significant items:

1. §3.0. Design criteria for structures, components, equipment and systems,
2. §6.0. Engineered safety features (containment systems), and
3. §15.0. Accident analyses.

Evaluation

We have reviewed this application for amendment submitted on December 19, 1980, and supplemented by letters of August 21, 1979, and December 30, 1980. Our review of safety-related matters and our conclusions concerning each item are described in the following subsections of this evaluation report.

Date: JUL 24 1981

§3.0 DESIGN CRITERIA FOR STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

We have reviewed the structural aspects pertaining to modifications of the enclosure building in relation to the primary containment integrity. We find that our prior reviews in Sections 3.3, 3.5, 3.7 and 3.8 of the Safety Evaluation Report are not affected by the removal of the enclosure building and its replacement with an equipment building from grade to the 270-foot level based on the following:

1. there is no change in the design methods and design criteria of the containment or other structures due to the enclosure building modifications. No structural credit has been taken for the enclosure building in either the original or modified design case;

2. the enclosure building modifications do not affect the plant's susceptibility to tornado missiles since it is assumed that the metal siding of the enclosure building afforded no protection from missiles;

3. the dynamic response characteristics of the containment building under seismic loads would essentially be unaltered because the enclosure building modifications result in only 2% reduction in weight and a very small change in stiffness of the overall containment building.

Based on the foregoing, we conclude that the safety margin of the original structural design or containment integrity is not significantly affected; therefore, the proposed design modification is acceptable.

JUL 24 1981

§6.0 ENGINEERED SAFETY FEATURES

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

In Supplement No. 6 to the Vogtle 1 & 2 PSAR, dated August 21, 1979, the Georgia Power Company revised its analysis of the radiological consequences of a postulated loss-of-coolant accident (LOCA) using assumptions that did not take credit for an enclosure building. This revised analysis is associated with the deletion of the enclosure building and its replacement with an equipment building from grade to the 270-foot level and the more restrictive containment leak rate of 0.2% per day.

Because the enclosure building was designed only to treat leakages from the containment building, the functional performance of the containment building in the event of a LOCA is not affected by the presence or absence of the enclosure building. We conclude that since the removal of the enclosure building will have no effect on our prior review of the containment system, the removal of the enclosure building is acceptable with respect to containment functional design.

§15.0 ACCIDENT ANALYSES

15.1 General

We have reviewed additional meteorological information submitted by the applicant on December 7, 1979, to determine the potential offsite doses calculated for the postulated loss-of-coolant accident (LOCA) and the fuel handling accident. The calculated doses are presented in Table 15.1 and the assumptions used are discussed in §15.2. A quantitative comparison of LOCA doses for the original and modified designs is presented in Table 15.2. All potential doses calculated by the applicant and by the staff for both the original and modified design cases are within the 10 CFR Part 100 guideline values. We conclude that the enclosure building modifications are acceptable.

TABLE 15.1

CALCULATED DOSES DUE TO DESIGN BASIS ACCIDENTS

	Exclusion Radius (1098 meters) 0-2 hours		Low Population Zone (3220 meters) 0-30 days	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-coolant (rem)	98	2.6	69	1.3
Fuel handling (rem)	8	0.7	1.3	0.2

JUL 24 1981

TABLE 15.2

COMPARISON OF LOCA DOSES FOR THE VOGTLE NUCLEAR STATION

	LOCA Doses per SER (March 1974) with Enclosure Building Primary Con. Leak Rate of 0.3%/day	LOCA Doses without Enclosure Building 0.2%/day Leak Rate & X/Q Values as of 1/80
Exclusion Area Boundary		
Thyroid (rem)	122	98
Whole body (rem)	7	2.6
Low Population Zone Boundary		
Thyroid (rem)	70	69
Whole body	8	1.3

JUL 24 1981

15.2 DESIGN BASIS ACCIDENT ASSUMPTIONS

15.2.1 Loss of Coolant Accident

We have reviewed the information supplied by the applicant in Supplement 8 to the PSAR. Based upon our review of this information, we have modelled the primary containment as two control volumes, consisting of the region covered by the containment spray and the remaining unsprayed containment volume. The spray region was assumed to be 78 percent of the total containment free volume of 2.62×10^{-6} cubic feet.

The staff also assumed that a 30 second delay exists from the initiation of the accident to the time spray injection into the containment begins. Further, the sprays, enhanced by the addition of sodium hydroxide, were assumed to operate until a reduction factor (DF) of 100 was reached in the sprayed region; after which the sprays did not remove any additional radioiodine. The single containment was assumed to leak at a design leak rate of 0.2% per day for the first 24 hours and at 0.1% per day thereafter.

The meteorological conditions and other important parameters used in our analysis of the consequences of a loss of coolant accident are tabulated in Table 15.3 and 15.4, and the calculated doses are given in Table 15.1.

JUL 24 1981

TABLE 15.3

ASSUMPTIONS USED TO ESTIMATE RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED LOSS OF COOLANT ACCIDENT AT VOGTLE UNITS 1 & 2

Power level, megawatts thermal	3565
Operating time, years	3
Primary containment leak rate, percent per day	0.2 to 24 hours 0.1 greater than 24 hours
Fraction of Core Inventory Available for Leakage from Containment, percent:	
Noble Gases	100
Iodine	25
Primary Containment Free Volume, cubic feet	2.75×10^{6}
Iodine form fractions, percent	
Elemental	91
Organic	4
Particulate	5
Spray removal rates, per hour	
Elemental	10
Particulate	0.45
Fraction of primary containment unsprayed, percent	22
Relative concentrations, second per cubic meter	
0-2 hours at 1060 meters	1.8×10^{-4}
0-8 hours at 3218 meters	3.3×10^{-5}
8-24 hours at 3218 meters	2.2×10^{-5}
24-96 hours at 3218 meters	9.2×10^{-6}
96-720 hours at 3218 meters	2.7×10^{-6}

JUL 24 1981

TABLE 15.4

Refueling Accident Calculation Input Parameters

Shutdown Time	100 hours
Total Number of Fuel Rods in the Core	50,952
Number of Fuel Rods Involved in the Refueling Accident	264
Power Peaking Factor	1.65
Iodine Fractions Released from Pool	
Elemental	75%
Organic	25%
Filter Efficiencies	
Elemental	90%
Organic	70%
X/Q Values, sec/m	
0-2 hours @ 1098 meters	1.8 x 10
0-2 hours @ 3220 meters	3.3 x 10

JUL 24 1981

Conclusion

We find the enclosure building modifications acceptable for the following reasons:

1) the safety margin of the original structural design or containment integrity is not significantly affected;

2) the functional performance of the containment building in the event of a LOCA is not affected by the presence or absence of the enclosure building; and

3) the calculated dose consequences are less than the guideline values of 10 CFR Part 100 in both the original and modified design cases.

We have concluded, based on the considerations discussed above, that with respect to the facility design changes authorized by Amendment Nos. 2 to Construction Permits CPPR-108 and CPPR-109, (1) there is reasonable assurance, taking into consideration the criteria contained in 10 CFR Part 100, that the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public, and (2) the issuance of these Amendments will not be inimical to the common defense and security or to the health and safety of the public.

Approval of this action does not preclude future changes as deemed necessary by accident analyses on the mitigation of more severe accident sequences than postulated design basis accidents which will be performed during the operating license review for the entire Vogtle plant.