Safety Evaluation Report

related to the preliminary design of the

GESSAR-238 Nuclear Island Standard Design

General Electric Company

Supplement No. 3

NUREG-0124
(Suppl. 3 to NUREG-75/110)

U. S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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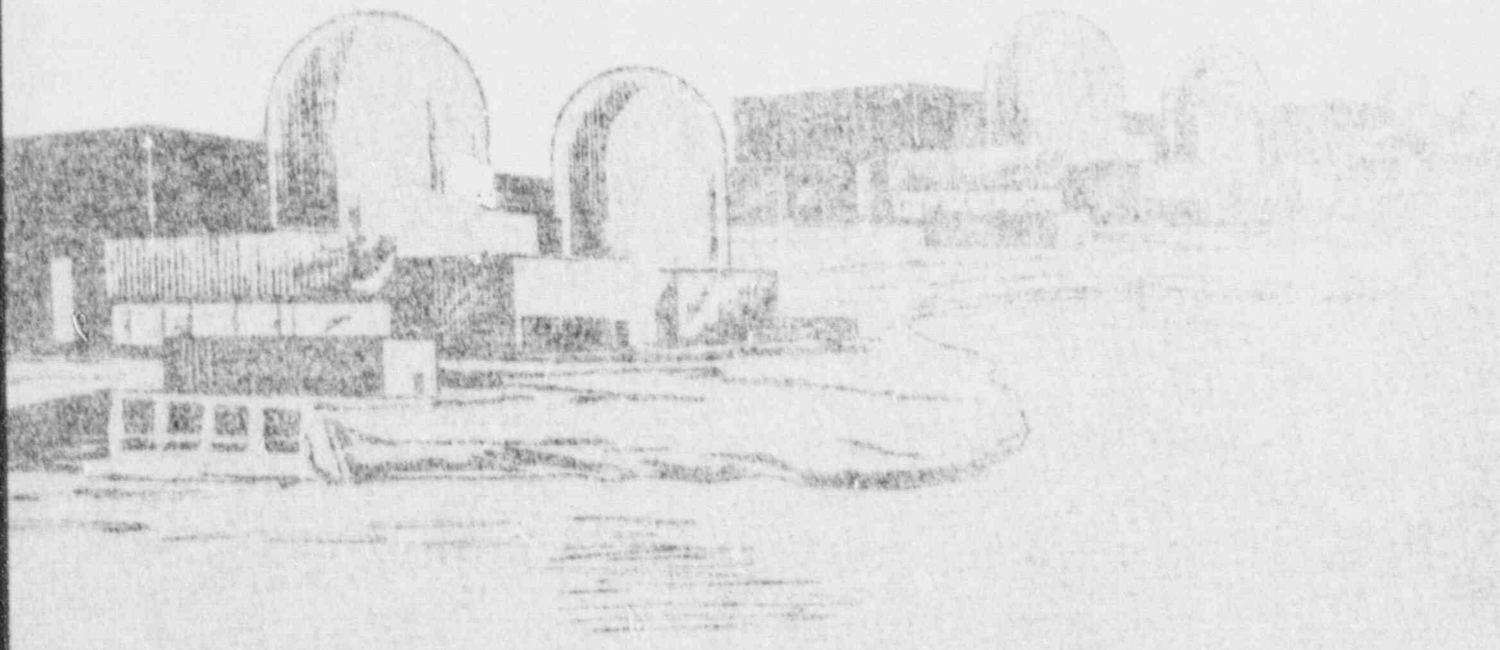

NUREG-0124 Suppl. 3
June 1977

SUPPLEMENT NO. 3
TO THE
SAFETY EVALUATION REPORT
BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION
IN THE MATTER OF
GENERAL ELECTRIC
STANDARD SAFETY ANALYSIS REPORT
(GESSAR-238 NUCLEAR ISLAND)
DOCKET NO. STN. 50-447

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction

On December 22, 1975, the United States Nuclear Regulatory Commission issued the Safety Evaluation Report (NUREG-75/110) and the Preliminary Design Approval for the General Electric Standard Safety Analysis Report (GESSAR-238 Nuclear Island) design (Docket No. STN 50-447). Supplement No. 1 to the Safety Evaluation Report, issued September 30, 1976, documented the resolution of several outstanding items, and summarized the status of those safety matters that remained to be resolved prior to a decision on the issuance of a construction permit for a referencing plant (Tables 1-1 and 1-2 of Supplement No. 1 to the Safety Evaluation Report). Supplement No. 2 to the Safety Evaluation Report, issued in February 1977, documented the resolution of all outstanding safety matters requiring resolution prior to a decision on the issuance of a construction permit for a referencing plant with the exception of two items - pool dynamics pipe loads in the region 17-19-5 feet above the suppression pool, and the new issue involving calculational errors in the emergency core cooling system performance evaluation.

The purpose of this supplement is to update our Safety Evaluation Report and Supplement Nos. 1 and 2 thereto by providing our evaluation of the additional information submitted by the General Electric Company to resolve these two remaining issues.

Each section of this supplement is numbered the same as the section of the Safety Evaluation Report, and is supplementary to and not in lieu of discussions in the Safety Evaluation Report, except where specifically so noted. Appendix A is a continuation of the chronology of our principal actions related to this application.

1.7 Facility Modifications as a Result of Regulatory Staff Review

1.7.2 Facility Modifications Required by the Staff

In Supplement No. 2 to the Safety Evaluation Report, we stated that the issue of pool dynamics pipe loads in the region of 17-19.5 feet above the suppression pool was still unresolved and that we would provide our evaluation of this matter in a future supplement.

Subsequent to the issuance of Supplement No. 2 to the Safety Evaluation Report, we provided our position on this matter to the General Electric Company in a letter from S. Varga (NRC) To G. Sherwood (General Electric) dated March 25, 1977. As discussed in Section 6.2.1.9 of this supplement, the General Electric has committed to meet this new position. We therefore consider this condition resolved.

1.8 Requirements for Future Technical Information

In Supplement No. 2 to the Safety Evaluation Report, we stated that the General Electric Company had informed us that certain calculational errors had been discovered which could affect the performance evaluation of the GESSAR-238 Nuclear Island emergency core cooling system.

Since the issuance of Supplement No. 2 to the Safety Evaluation Report, the General Electric Company has incorporated the necessary corrections and submitted a loss-of-coolant accident reanalyis and reported the results in letters from A. Levine, General Electric Company to D. Ross, Jr., NRC dated February 14, 1977 and C. Fuller, General Electric Company to D. Vassallo, NRC, dated February 17, 1977.

As discussed in Section 6.3.2 of this supplement, we have reviewed these submittal and conclude that they are acceptable.

1.11 Conclusion

In Supplement No. 2 we concluded that the General Electric Company had provided sufficient information, on the GESSAR-238 Nuclear Island portion of the plant's design, to provide suitable bases for the issuance of a construction permit to a referencing plant, with the exception of the following two issues:

(1) Pool dynamics pipe loads in the region 17-19.5 feet above the suppression pool, and

(2) Calculational errors in the emergency core cooling system performance evaluation.

As discussed in Sections 6.2.1.9 and 6.3.2 of this supplement, the General Electric Company has since provided additional information on these matters which has been reviewed by the staff and found acceptable.

Therefore, with the issuance of this supplement to the Safety Evaluation Report and Amendment No. 1 to PDA-1, there exists no open licensing issues on the GESSAR-238 Nuclear Island application requiring resolution prior to a decision on the issuance of a construction permit to an applicant referencing the GESSAR-238 Nuclear Island design.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.9 Containment Pool Dynamics

Our position on loss-of-coolant accident pool swell loads was given in Section 6.2.1.9 of the Safety Evaluation Report. As was noted in that section, the General Electric Company took exception to our criteria in the 17-19.5 foot region. Therefore, our criteria in this region was made a condition of the Preliminary Design Approval.

Subsequent to the issuance of PDA-1, the staff initiated a reevaluation of the suppression pool impact load criteria for beams and pipes in the transition zone (17-19.5 feet) for Mark III containments. After carefully considering the findings of our consultants and our own evaluation, we conclude that the transition impact load criteria should be modified as shown in Figure 6.2-1 of this supplement.

Figure 6.2-1 contains both the previous and revised criteria. In the revised criteria the pressure load has been increased significantly for beams in a portion of the transition region and reduced for pipes. These changes result from:

(1) Increases in the predicted maximum pool swell velocity.

(2) Increases in the height above the pool at which the slug thickness is one to two feet.

(3) Our assessment of the impact characteristics of the pool slug in the one to two-foot range.

In an April 8, 1977 letter from W. Gilbert (General Electric) to S. Varga (NRC), the General Electric Company adopted these new loading criteria as one of the design bases for their GESSAR-238 Nuclear Island design. We therefore consider the pool dynamics condition on PDA-1 to be resolved.

6.3 Emergency Core Cooling System

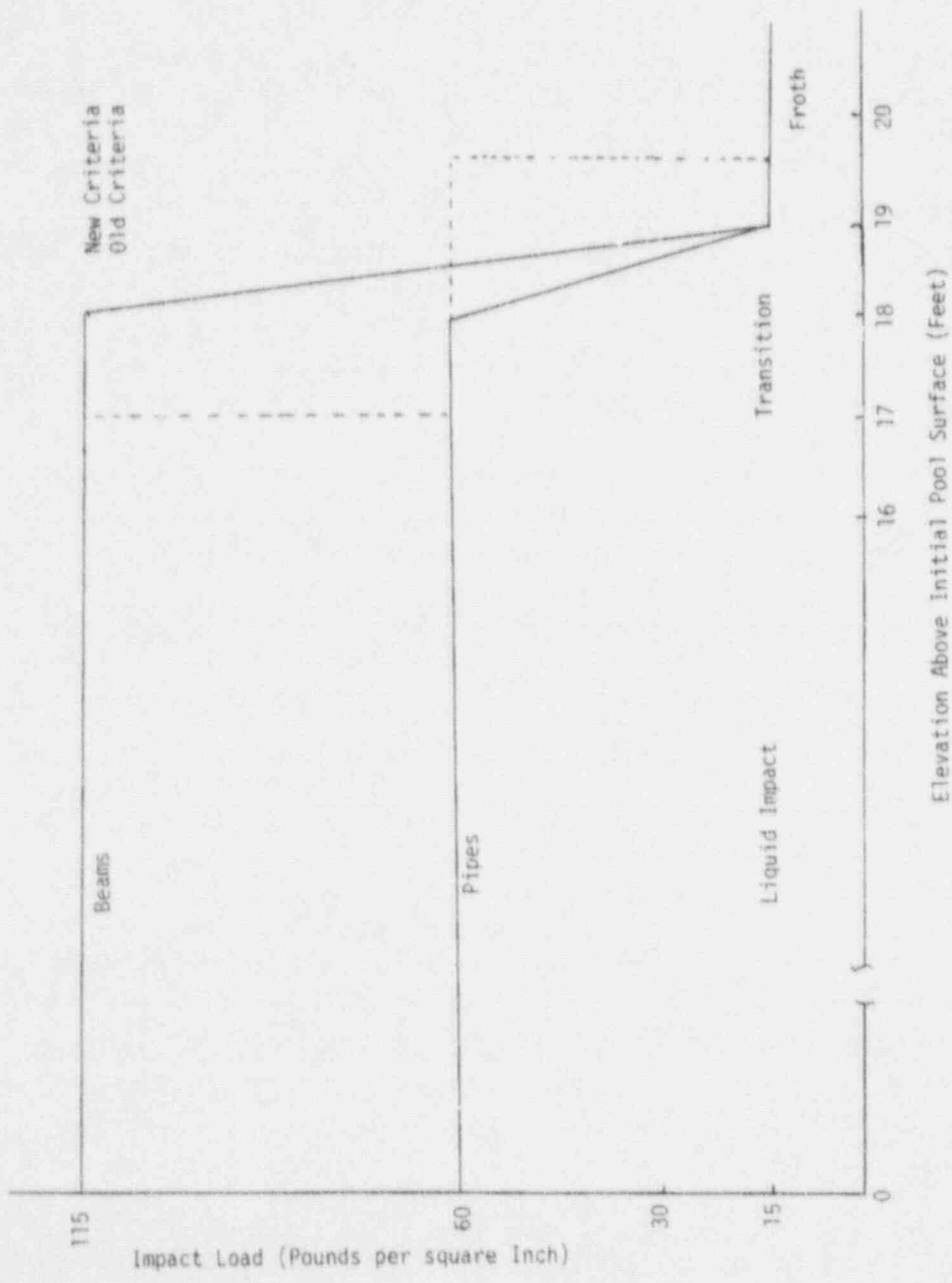
6.3.2 Performance Evaluation

As discussed in Supplement No. 2 to the Safety Evaluation Report, the staff was informed by the General Electric Company that certain calculational errors had been discovered during the General Electric Company reverification program which could affect the performance evaluation of the GESSAR-238 Nuclear Island emergency core cooling system. We have received General Electric Company letters (E. D. Fuller,

Figure 6.2-1

POOL SWELL

IMPACT LOAD CRITERIA



General Electric, to D. F. Ross Jr., NRC) dated January 19, 1977; A. J. Levine to D. B. Vassallo, NRC, dated January 21, 31, 1977; and A. J. Levine to D. F. Ross, Jr., dated February 7, 1977) which describe the calculational changes made for GESSAR-238 Nuclear Island emergency core cooling system performance evaluation along with revised analyses. The calculational changes were identified as:

(1) Correction of the water level setpoint in the SAFE code for high pressure core spray system initiation.

(2) Modification of the core power in the REFLOOD code.

(3) Correction in the design basis accident break area.

(4) Correction of the guide tube thermal resistance.

(5) Correction of the initial core fluid quality.

(6) Correction of the vaporization calculational constants in the REFLOOD code.

By letters from A. Levine, General Electric Company, to D. Ross, Jr., dated October 13, 1976 and January 20, 1977, the General Electric Company also requested a change in the REFLOOD code. The staff has reviewed this request and finds it acceptable.

In addition, the staff concluded that the General Electric Company incorrectly modeled counter-current flow-limiting effects on the emergency core cooling system water entering the core. Counter-current flow-limiting effects within the fuel channels raises the pressure differential between the upper and lower plena compared to the pressure differential that would be predicted if only single phase steam flow is assumed in the channels. This added pressure differential from the two-phase flow consideration tends to reduce the flow of spray water between the bypass region and the lower plenum.

The General Electric Company REFLOOD computer programs did not account for the two-phase pressure drop in the fuel bundle; consequently, they predict nonconservative times for core reflood. The staff has reviewed available information on counter-current two-phase pressure drops in pipes for representative hydraulic diameters. Based on this information, we required that the General Electric Company include a one pound per square inch correction in the calculation of driving head to predict bypass flow to the lower plenum until more experimental evidence is available to support an appropriate model.

After incorporating all corrections, the General Electric Company submitted a loss-of-coolant accident reanalysis and reported the results in letters from A. Levine, General Electric Company to D. Ross, Jr., NRC, dated February 14, 1977 and E. Fuller, General Electric Company to D. Vassallo, NRC, dated Feburary 17, 1977.

The results of the calculations performed in accordance with Appendix K to 10 CFR Part 50 for GESSAR-238 Nuclear Island (based on a maximum average planar linear heat generation rate of 12.27 kilowatts per foot) show a peak cladding temperature of 2038 degrees Fahrenheit; a peak local oxidation of less than two percent; and a ma⁻⁻ um core average hydrogen generation of less than 0.14 percent for the worst break assuming a failure of the low pressure coolant injection system diesel. The previous break spectrum submitted in May 1975 for GESSAR-238 Nuclear Island was based on unity local peaking factors for all rods in the limiting bundle. For this previous analysis, the maximum linear heat generation rate selected was 13.4 kilowatts per foot (this can be interpreted as a maximum average planar heat generation rate of 13.4 kilowatts per foot). Using a maximum average planar linear heat generation rate of 12.27 kilowatts per foot, based on conservative exposure dependent local peaking factors, produces a break spectrum of the same general shape but of lower magnitude than previously submitted, with the largest break size yielding the highest peak cladding temperature.

We have reviewed the analysis of the emergency core cooling system performance submitted by the General Electric Company for GESSAR-238 Nuclear Island and conclude that the analysis performed is in conformance with the requirements of 10 CFR Part 50, Section 50.46(a). The GESSAR-238 Nuclear Island emergency core cooling system performance assures conformance with (1) the peak cladding temperature limit of 2200 degrees Fahrenheit, (2) the maximum cladding oxidation limit of 17 percent of total cladding thickness before oxidation, (3) the maximum hydrogen generation core-wide limit of one percent of the total metal in the cladding, (4) the core geometry remaining amenable to cooling, and (5) the long term cooling requirement of maintaining acceptable core temperatures and decay heat removal.

During our review of the analyses of BWR/5 and BWR/6 reactors, we expressed a concern relating to recirculation flow control valve closure in the event of a design basis loss-of-coolant accident. The results of a General Electric Company sensitivity study to evaluate the effects of fast closure of a recirculation flow control valve concident with the design basis loss-of-coolant accident and worst postulated emergency core cooling system failure were submitted in a letter dated April 25, 1975 form A. Levine to V. Stello of the NRC staff. The results of this sensitivity study show that the calculated peak cladding temperature remains below 2200 degrees Fahrenheit.

With regard to the emergency core cooling system reanalysis with the changes noted above, the staff did not require the sensitivity study related to recirculation flow control valve closure to be reanalyzed. The General Electric Company has stated that: (1) the valve will fail "as is" with loss of offsite power - this is, in part, due to a spring-loaded valve design feature which locks the valve in the "as is" position in the event of loss of hydraulic pressure for any reason; (2) the valve is designed to close only to a position which permits 30 percent flow; and (3) during a loss-of-coolant accident the high containment pressure signal shifts the recirculation flow control valve to the manual mode from the automatic mode. The General Electric Company has also stated that all electrical systems for the recirculation flow control valve operation are outside the containment and thus are not subject to the loss-of-coolant accident environment. The previous sensitivity study of fast closure of the recirculation flow control valve (100 percent closure) coincident with a design basis loss-of-coolant accident and a worst-case postulated emergency core cooling system failure showed a peak cladding temperature increase of less than 100 degrees Fahrenheit. This design feature is being reviewed by the staff on the Wm. H. Zimmer Nuclear Power Station, Unit 1, Docket No. 50-358 at the final design stage. Even if a revised analyses with postulated recirculation flow control valve closure yields a peak cladding temperature greater than 2200 degrees Fahrenheit, or if any of the other cirteria of Section 50.46 of 10 CFR Part 50 are exceeded, a reduction in the permissible value of the maximum average planar linear hect generation rate can be included in the technical specifications to mitigate postulated recirculation flow control valve effects.

We conclude that the probability of valve motion as a consequence of a loss-of-coolant accident is small; however, any change required in the valve design or emergency core cooling system analysis because of recirculation valve motion can be implemented as part of the final design stage of review.

In summary, we conclude that the performance evaluation emergency core cooling system for GESSAR-238 Nuclear Island meet all the criteria of Section 50.46 of 10 CFR Part 50 and the requirements of Appendix K to 10 CFR 50 and is acceptable.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW

February 11, 1976	The Nuclear Regulatory Commission issued Supplement No. 2 to the GESSAR-238 Nuclear Island Safety Evaluation Report.
February 14, 1977	Letter from A. Levine of General Electric to D. Ross, Jr., of NRC containing information on the emergency core cooling system performance evaluation.
February 17, 1977	Letter from E. Fuller of General Electric to D. Vassallo of NRC containing information on the emergency core cooling system performance evaluation.
February 17, 1977	Letter from G. Sherwood of General Electric to B. Rusche of NRC requesting our position on pool swell loads.
March 25, 1977	Letter from S. Varga of NRC to G. Sherwood of General Electric transmitting our new pool swell loads to General Electric.
April 8, 1977	Letter from W. Gilbert of General Electric to S. Varga of NRC adopting our pool swell loads as a design basis for the GESSAR-238 Nuclear Island design.