Safety Evaluation Report

NUREG-0052
(Suppl. 2 to NUREG 75/082)

U. S. Nuclear
Regulatory Commission

related to construction of

Sterling Power Project Nuclear Unit No. 1

Office of Nuclear
Reactor Regulation

Docket No. STN 50-485

Rochester Gas and Electric Corporation, et al.

December 1976

Supplement No. 2

Available from
National Technical Information Service
Springfield, Virginia 22161
Price: Printed Copy $3.50 ; Microfiche $3.00

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December 1, 1976

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

ROCHESTER GAS AND ELECTRIC CORPORATION

ET AL

STERLING POWER PROJECT NUCLEAR UNIT NO. 1

DOCKET NO. STN 50-485

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission's (Commission) Safety Evaluation Report in the matter of the application by the Rochester Gas and Electric Corporation, the Orange and Rockland Utilities, Incorporated, the Niagara Mohawk Power Corporation and the Central Hudson Gas and Electric Corporation (applicants) to construct and operate the proposed Sterling Power Project Nuclear Unit No. 1 was issued on September 5, 1975. Supplement No. 1 to the Safety Evaluation Report was issued on April 14, 1976. We indicated in Supplement No. 1 that a favorable resolution of each of the outstanding issues had been made and that our safety evaluation of the application had been completed.

The purpose of this supplement is to update the Safety Evaluation Report (and Supplement No. 1) by providing our evaluation of additional information submitted by the applicants since the issuance of Supplement No. 1 concerning a reanalysis of the emergency core cooling system.

Each of the sections in this supplement is numbered the same as the section of the Safety Evaluation Report and Supplement No. 1 that is being updated and is supplementary to and not in lieu of the discussion of the Safety Evaluation Report and Supplement No. 1. Appendix A to this supplement is a continuation of the chronology of our principal actions related to the processing of the application.

6.0 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

In Section 6.3.5 of Supplement No. 1 to the Safety Evaluation Report, we concluded that the design of the Sterling emergency core cooling system complies with the Final Acceptance Criteria.

In a letter dated August 13, 1976, Westinghouse Electric Corporation reported to us that measurements made in an operating plant and associated calculations have indicated that the temperature of the reactor coolant in the upper head region of the reactor vessel may be higher than the temperature which was assumed in the emergency core cooling system analyses performed for Westinghouse two, three and four loop plants. Westinghouse has performed sensitivity studies that show the peak clad temperature of the fuel element for a postulated loss-of-coolant accident increases with an increase in the upper head coolant temperature. Since the analysis of the emergency core cooling system for all of the SNUPPS plants was performed with a lower coolant temperature for the upper head region, we requested the applicants to perform a reanalysis of the SNUPPS plants, which conservatively assumes an upper head coolant temperature equal to the hot leg coolant temperature, to reaffirm that the emergency core cooling system design for these plants can still comply with the Final Acceptance Criteria.

In a letter dated October 15, 1976, the applicants submitted a loss-of-coolant accident analysis for four postulated large pipe ruptures using the hot leg coolant temperature in the upper head region. The Westinghouse Topical Report WCAP-8865 (October 1976), "Westinghouse ECCS - Four-Loop Plant (17 x 17) Sensitivity Studies," which included the appropriate generic break study that used the increased upper head coolant temperature, was referenced by the applicants. This generic study indicated that the double-ended cold leg guillotine rupture was still the most limiting break for four-loop plants. These calculations satisfy the break spectrum requirements of Section 50.46 of 10 CFR Part 50.

The analyses were performed with a modified version of the Westinghouse evaluation model used for the previous analyses reported in Section 6.3.3 of Supplement No. 1 to the Safety Evaluation Report. This modified model was also found to be acceptable as documented in the Commission's letter to Westinghouse dated May 13, 1976.

The analysis to determine the containment backpressure for the reanalysis of the SNUPPS emergency core cooling system was performed in the same way as the previous analysis discussed in Section 6.2.1 of the Safety Evaluation Report. This analysis

was performed with a reference containment using assumptions which meet the requirements defined in Section 6.2.1 of the Safety Evaluation Report. The resultant containment backpressure from this analysis was then used in the reanalysis of the emergency core cooling system.

The applicants also performed a containment backpressure analysis using best estimate parameters for a typical SNUPPS containment. This analysis showed that higher containment backpressures would occur than for the reference containment. Higher containment backpressures will result in lower peak clad temperatures in the analysis of emergency core cooling systems. Therefore, we conclude that the applicants have shown that the reference containment analysis is conservative for the reanalysis of the emergency core cooling system for the SNUPPS plants.

We reaffirm our conclusion, stated in Section 6.2.1 of the Safety Evaluation Report, that the plant-dependent information used for the emergency core cooling system containment pressure analysis for the SNUPPS plants is conservative, and that the calculated containment pressure is in accordance with Appendix K to 10 CFR Part 50.

The new analyses identified the worst break as a double-ended cold leg break with a discharge coefficient (Moody multiplier) of 1.0. The peak clad temperature of the fuel element was calculated to be 2148 degrees Fahrenheit, which is below the acceptable limit of 2200 degrees Fahrenheit as specified in Section 50.46 of 10 CFR Part 50. In addition, the calculated maximum local metal-water reaction of 6.7 percent and a total core-wide metal-water reaction of less than 0.3 percent are well below the allowable limits of 17 percent and one percent, respectively. These analyses were performed using a total peaking factor of 2.32, 102 percent of the rated nuclear steam supply system power level of 3411 megawatts thermal and 102 percent of the peak linear power density of 12.6 kilowatts per foot. The analyses also conservatively assumed the loss of offsite power and the loss of one low head safety injection pump.

On the basis of this evaluation and our previous evaluation described in Section 6.3.3 of Supplement No. 1 to the Safety Evaluation Report, we conclude that the emergency core cooling system performance for the SNUPPS plants conforms to the acceptance criteria in Section 50.46 of 10 CFR Part 50. Therefore, we reaffirm our conclusion that the design of the Sterling emergency core cooling system complies with the Final Acceptance Criteria.

21.0 CONCLUSIONS

Our evaluation of the reanalysis of the emergency core cooling system has confirmed that the system still complies with the Final Acceptance Criteria. Hence, in Supplement No. 1 and in this supplement, we have discussed each of the outstanding issues identified in Section 1.8 of the Safety Evaluation Report and have indicated a favorable resolution of each matter. Therefore, we reaffirm our conclusions as set forth in Section 21.0 of the Safety Evaluation Report.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW OF STERLING PLANT

April 14, 1976	Issuance of Supplement No. 1 to Safety Evaluation Report.
May 3, 1976	Letter from SNUPPS concerning use of ASME Code Cases.
May 3, 1976	Letter to applicants advising of revision to Standard Review Plan on fire protection criteria.
May 6, 1976	Letter to applicants advising of new filing requirements.
May 10, 1976	Submittal of Amendment No. 37 (Revision to General Information of Application).
June 28, 1976	Letter to applicants concerning use of ASME Code Cases, in response to SNUPPS letter of May 3, 1976.
July 2, 1976	Letter to applicants concerning ATWS.
July 8, 1976	Meeting with SNUPPS to discuss submittals of Final Safety Analysis Reports and Post Construction Permit revisions to Preliminary Safety Analysis Reports.
August 9, 1976	Letter to applicants changing copy requirements for Safety Analysis Reports, Amendments and Environmental Reports.
August 25, 1976	Letter to applicants requesting reanalysis of ECCS.
September 3, 1976	Letter from SNUPPS transmitting, "Evaluation of the Effect of Assuming Upper Head Hot Leg Temperature in ECCS Analysis - SNUPPS," in response to request of August 25, 1976.
September 3, 1976	Letter from applicants incorporating SNUPPS letter of September 3, 1976.
September 8, 1976	Meeting with SNUPPS to discuss fire protection criteria and steam tunnel design criteria.
September 23, 1976	Letter to applicants requesting additional information concerning ECCS analyses.

September 28, 1976	Letter from SNUPPS providing submittal date for additional information concerning ECCS analysis.
September 30, 1976	Letter to applicants requesting information concerning fire protection evaluation.
October 15, 1976	Letter from SNUPPS attaching a copy of a document entitled, "Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)", in response to request of September 23, 1976.
October 18, 1976	Letter from applicants incorporating SNUPPS letter of October 15, 1976.
October 22, 1976	Letter from applicants advising that fire protection information will be submitted on April 1, 1977, in response to request of September 30, 1976.