TECHNICAL EVALUATION REPORT

PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION (B-69)

ROCHESTER GAS AND ELECTRIC COMPANY

ROBERT E. GINNA NUCLEAR POWER PLANT

NRC DOCKET NO. 50-244

NRC TAC NO. 46835

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 5

FRC TASK 146

Prepared by

Franklin Research Center
20th and Race Street
Philadelphia, PA 19103

Author: F. W. Vosbury

FRC Group Leader: R. C. Herrick

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: P. Hearn

December 15, 1982

Prepared by:	Reviewed by:	Approved by:
FW Vosbury	R C Herrick	D F Carfagno
Principal Author	Group Leader	Department Director
Date: 12-6-82	Date: 12/14/82	Date: 12-15-82

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

CONTENTS

Franklin Research Center
A Division of The Franklin Institute

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. F. W. Vosbury contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents the review of Rochester Gas and Electric Corporation's (RG&E) response to the Nuclear Regulatory Commission's (NRC) IE Bulletin 80-04, "Analysis of a Pressurized Water Reactor Main Steam Line Break with Continued Feedwater Addition" [1], as it pertains to the Robert E. Ginna Nuclear Power Plant. This evaluation was performed with the following objectives:

- o to assess the conformance of RG&E's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess RG&E's proposed interim and long-range corrective action plans and schedules, if needed, as a result of the MSLB analyses.

1.2 GENERIC BACKGROUND

In the summer of 1979, a pressurized water reactor (PWR) licensee submitted a report to the NRC that identified a deficiency in the plant's original analysis of the containment pressurization resulting from a MSLB. A reanalysis of the containment pressure response following a MSLB was performed, and it was determined that, if the auxiliary feedwater (AFW) system continued to supply feedwater at runout conditions to the steam generator that had experienced the steam line break, containment design pressure would be exceeded in approximately 10 minutes. The long-term blowdown of the water supplied by the AFW system had not been considered in the earlier analysis.

On October 1, 1979, the foregoing information was provided to all holders of operating licenses and construction permits as IE Information Notice 79-24 [2]. Another facility performed an accident analysis review pursuant to receipt of the information in the notice and discovered that, with offsite electrical power available, the condensate pumps would feed the affected steam generator at an excessive rate. This excessive feed was not previously considered in the plant's analysis of a MSLB accident.

A third licensee informed the NRC of an error in the MSLB analysis for their plant. During a review of the MSLB analysis, for zero or low power at the end of core life, the licensee identified an incorrect postulation that the startup feedwater control valves would remain positioned "as is" during the transient. In reality, the startup feedwater control valves will ramp to 80% full open due to an override signal resulting from the low steam generator pressure reactor trip signal. Reanalysis of the events showed that opening of the startup valve and associated high feedwater addition to the affected steam generator would cause a rapid reactor cooldown and resultant reactor return-to-power response, a condition which is outside the plant design basis.

Because of these deficiencies identified in original MSLB accident analyses, the NRC issued IE Bulletin 80-04 on February 8, 1980. This bulletin required all PWRs with operating licenses and certain near-term PWR operating license applicants to perform the following:

"1. Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

2. Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

   a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

   b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

3. If the potential for containment overpressure exists or the reactor return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed."

1.3 PLANT-SPECIFIC BACKGROUND

Rochester Gas and Electric Corporation responded to IE Bulletin 80-04 in a letter to the NRC dated April 30, 1980 [3]; additional information was obtained from an NRC letter dated November 3, 1981 [4] and RG&E's response dated February 1, 1982 [5] concerning the Systematic Evaluation Program (SEP) topics VI-2.D, "Mass and Energy Release for Possible Pipe Break Inside Containment," and VI-3, "Containment Pressure and Heat Removal Capability," for the Ginna Plant. The information in References 3, 4, and 5 has been evaluated along with pertinent information from the R. E. Ginna Nuclear Power Plant Final Facility Description and Safety Analysis Report (FFDSAR) [6] to determine the adequacy of the Licensee's compliance with IE Bulletin 80-04.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [7]:

1. PWR licensees' responses to IE Bulletin 80-04 shall include the following information related to their analysis of containment pressure and core reactivity response to a MSLB within or outside containment:

   a. A discussion of the continuation of flow to the affected steam generator, including the impact of runout flow from the AFW system and the impact of other energy sources, such as continuation of feedwater or condensate flow. AFW system runout flow should be determined from the manufacturer's pump curves at no backpressure, unless the system contains reliable anti-runout provisions or a more representative backpressure has been conservatively calculated. If a licensee assumes credit for anti-runout provisions, then justification and/or documentation used to determine that the provisions are reliable should be provided. Examples of devices for which provisions are reliable are anti-runout devices that use active components (e.g., automatically throttled valves) which meet the requirements of IEEE Std 279-1971 [8] and passive devices (e.g., flow orifices or cavitating venturis).

   b. A determination of potential containment overpressure as a result of the impact of runout flow from the AFW system or the impact of other energy sources such as continuation of feedwater or condensate flow. Where a revised analysis is submitted or where reference is made to the existing FSAR analysis, the analysis must show that runout AFW flow was included and that design containment pressure was not exceeded.

   c. A discussion of the ability to detect and isolate the damaged steam generator from continued feedwater addition during the MSLB accident. Operator action to isolate AFW flow to the affected steam generator within the first 30 minutes of the start of the MSLB should be justified. Where operator action is to be completed within the first 10 minutes, then the justification should address the indication available to the operator and the actions required. Where operator action is required to prevent exceeding a design value, i.e., containment design pressure or specified acceptable fuel design limits, then the discussion should include the calculated time when the design value would be exceeded if no operator action were assumed. Where operator actions are to be performed between 10 and 30 minutes after the start of the MSLB, the justification should address the indications available to the operator and the operator actions required, noting that for the first 30 minutes, all actions should be performed from the control room.

d. Where all water sources were not considered in the previous analysis, an indication should be provided of the core reactivity change which results from the inclusion of additional water sources. A submittal which does not determine the magnitude of reactivity change from an original analysis is not responsive to the requirements of IE Bulletin 80-04.

2. If containment overpressure or a worsening of the reactor return-to-power with a violation of the specified acceptable fuel design limits described in Section 4.2 of the Standard Review Plan [9] (i.e., increase in core reactivity) can occur by the licensee's analysis, the licensee shall provide the following additional information:

   a. the proposed corrective actions to prevent containment over-pressure or the violation of fuel design limits and the schedule for their completion

   b. the interim actions that will be taken until the proposed corrective action is completed, if the unit is operating.

3. The acceptable input assumptions used in the licensee's analysis of the core reactivity changes during a MSLB are given in Section 15.1.5 of the Standard Review Plan [10]. The following specific assumptions should be used unless the analysis shows that a different assumption is more limiting:

Assumption II.3.b.:	Analysis should be performed to determine the most conservative assumption with respect to a loss of electrical power. A reactivity analysis should be conducted for a normal power situation as well as a loss of offsite power scenario, unless the licensee has previously conducted a sensitivity analysis which demonstrates that a particular assumption is more conservative.
Assumption II.3.d.:	The most restrictive single active failure in the safety injection system which has the effect of delaying the delivery of high concentration boric acid solution to the reactor coolant system, or any other single active failure affecting the plant response, should be considered.

Assumption II.3.g.: The initial core flow should be chosen such that the post-MSLB shutdown margin is minimized (i.e., maximum initial core flow).

The acceptable computer codes for the licensee's analysis of core reactivity changes are, by nuclear steam supply system (NSSS) vendor, the following: CESEC (Combustion Engineering), LOFTRAN (Westinghouse), and TRAP (Babcock & Wilcox). Other computer codes may be used, provided that these codes have previously been reviewed and found to be acceptable by the NRC staff. If a computer code is used which has not been reviewed, the licensee must describe the method employed to verify the code results in sufficient detail to permit the code to be reviewed for acceptability.

4. If the AFW pumps can be damaged by extended operation at runout flow, the licensee's action to preclude damage should be reviewed for technical merit. Any active features should satisfy the requirements of IEEE Std 279-1971. Where no corrective action has been proposed, this should be indicated to the NRC for further action and resolution.

5. Modifications to electrical instrumentation and controls needed to detect and initiate isolation of the affected steam generator and feedwater sources in order to prevent containment overpressure and/or unacceptable core reactivity increases must satisfy safety-grade requirements. Instrumentation that the operator relies upon to follow the accident and to determine isolation of the affected steam generator and feedwater sources should conform to the criteria contained in ANS/ANSI-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors" [11], and the regulatory positions in Regulatory Guide 1.97, Rev. 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [12].

6. AFW system status should be reviewed to ensure that system heat removal capacity does not decrease below the minimum required level as a result of isolation of the affected steam generator and also that recent changes have not been made in the system which adversely affect vital assumptions of the containment pressure and core reactivity response analyses.

7. The safety-grade requirements (redundancy, seismic and environmental qualifications, etc.) of the equipment that isolates the main feedwater (MFW) and AFW systems from the affected steam generator should be specified. The modifications of equipment that is relied upon to isolate the MFW and AFW systems from the affected steam generator should satisfy the following criteria to be considered safety-grade:

- o Redundancy and power source requirements: The isolation valves should be designed to accommodate a single failure. A failure-modes-and-effects analysis should demonstrate that the system is capable of withstanding a single failure without loss of function. The single failure analysis should be conducted in accordance with the appropriate rules of application of ANS-51.7/N658-1976, "Single Failure Criteria for PWR Fluid Systems" [13].
- o Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [14].
- o Environmental qualification: The isolation valves should satisfy the requirements of NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" [15].
- o Quality standards: The isolation valves should satisfy Group B quality standards as recommended in Regulatory Guide 1.26 or similar quality standards from the plant's licensing bases.

3. TECHNICAL EVALUATION

The scope of work included the following:

1. Review the Licensee's response to IE Bulletin 80-04 against the acceptance criteria.

2. a. Evaluate the Licensee's MSLB analyses for the potential of overpressurizing the containment and with respect to the core reactivity increase due to the effect of continued feedwater flow.

   b. Evaluate the Licensee's proposed corrective actions and schedule for implementation if the findings of Task 2a indicate that a potential exists for overpressurizing the containment or worsening the reactor return-to-power in the event of a MSLB accident.

3. Prepare a TER for each plant based on the evaluation of the information presented for Tasks 1 and 2 above.

This report constitutes a TER in satisfaction of Task 3. Sections 3.1 through 3.3 of this report state the requirements of IE Bulletin 80-04 by subsection, summarize the Licensee's statements and conclusions regarding these requirements, and present a discussion of the Licensee's evaluation followed by conclusions and recommendations.

3.1 REVIEW OF CONTAINMENT PRESSURE RESPONSE ANALYSIS

The requirement from IE Bulletin 80-04, Item 1, is as follows:

"Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow."

3.1.1 Summary of Licensee Statements and Conclusions

In regard to the review of the containment pressure response analysis, the Licensee stated [3]:

"Although the Ginna post-steamline break containment pressure analysis in the FSAR did not include the effects of auxiliary feedwater flow to the affected steam generator, it is important to recognize that the evaluation also did not include the benefits of passive and active heat sinks inside containment. Continued feedwater/condensate addition to the steam generator will not occur, since the safety injection signal (generated by a variety of process parameters, including high steam line flow, high containment pressure, and low pressurizer pressure) will close the feedwater control valves and stop the feedwater pumps. The addition of maximum auxiliary feedwater flow to the broken steam generator will eventually require operator action to 1) realign flow to the intact generator, 2) terminate auxiliary feedwater flow to the broken generators. Positive information is available to the operator to determine which is the affected steam generator. Steam generator level instrumentation is located inside containment and steam generator pressure is located outside containment where it would not be affected by the accident environment inside containment. It is expected that, through proper training and by use of the emergency procedures, the operators will be capable of quickly recognizing the steam line break, and will perform the proper operations.

There is substantial time available for the operator to perform the two safety functions noted above. The SEP Safe Shutdown review concluded following their site visit in June 1978 that one steam generator would not boil dry for over thirty minutes. Thus there is substantial time to align flow to the intact steam generator. The termination of auxiliary feedwater flow to the affected steam generator, under the pessimistic circumstances, would require more rapid action (but still easily within the capability of the operators) to maintain containment pressure below design pressure. The analysis presented in Attachment 1 concludes that, assuming minimum safeguards for containment cooling, auxiliary feedwater flow would have to be terminated in about 26 minutes. With maximum safeguards, this time would be extended to about 44 minutes."

In regard to considering AFW runout flow in the containment analysis, the Licensee stated [3]:

"A potential single failure of the flow controller to control flow to 200 gpm is not considered a worst-case single failure in terms of net energy addition to the containment, since the operation of all containment cooling safeguards (vs. the minimum safeguards assumed in this evaluation) would result in a substantial increase in energy removal from containment."

In regard to the ability of the AFW pumps to remain operable during a MSLB, the Licensee stated [3]:

"There is no need to consider the operation of the auxiliary feedwater pumps at runout flow. The turbine-driven pumps are controlled by a governor, and will not exceed about 400 gpm. The motor driven pump flow is controlled by the AFW control valves, which receive an automatic throttle signal to 200 gpm from their flow controllers."

3.1.2 Evaluation

The Licensee's submittals [3, 4, 5] concerning the containment pressure response following a MSLB and applicable sections of the Ginna Nuclear Power Plant FFDSAR [6] were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

- Criterion 1.a - Continuation of flow to the affected steam generator
- Criterion 1.b - Potential for containment overpressure
- Criterion 1.c - Ability to detect and isolate the damaged steam generator
- Criterion 4 - Potential for AFW pump damage
- Criterion 5 - Design of steam and feedwater isolation system
- Criterion 6 - Decay heat removal capacity
- Criterion 7 - Safety-grade requirements for MFW and AFW isolation valves.

The Ginna Nuclear Power Plant is a Westinghouse-designed, 2-loop, 1520-MWt plant.

The following systems provide the necessary protection against a steam pipe rupture:

- Safety injection system actuation on:
  - a. two out of three pressurizer low pressure signals
  - b. two out of three low pressure signals in any steam line
  - c. two out of three high containment pressure signals
  - d. high steam flow.
- The overpower reactor trips (nuclear flux and differential temperature) and the reactor trip occurring upon actuation of the safety injection system.

- o Redundant isolation of the MFW lines. A safety injection signal will rapidly close all MFW control valves, trip the main MFW pumps, and close the MFW pump discharge valves. In addition, normal control action will also signal the MFW valves to close.

- o Trip of the fast-acting steam line isolation valves (designed to close in less than 5 seconds with no flow) on:

  a. one out of two high steam flow signals in a steam line in coincidence with any safety injection signal. (Dual setpoints are provided, with the lower setpoint used in coincidence with two out of four indications of low reactor coolant average temperature)

  b. two out of three high containment pressure signals.

Each steam line has a fast-closing isolation valve and a check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For a break upstream of the isolation valve in one line, closure of either the check valve in that line or the isolation valve in the other line will prevent blowdown of the other steam generator.

The AFW system consists of two redundant systems, the main AFW system and the standby AFW system. The main AFW system includes one turbine-driven pump (400 gpm) and two motor-driven pumps (200 gpm each) and is automatically actuated on the receipt of the following signals.

<u>Motor-driven AFW pumps</u>

- o low-low steam generator level (two out of three channels on either steam generator)
- o trip of both MFW pumps
- o safety injection signal

<u>Turbine-driven AFW pump</u>

- o low-low steam generator level (two out of three channels on both steam generators)
- o loss of voltage on both 4-kV buses.

The main motor-driven AFW pumps are normally aligned so that each pump supplies one steam generator. The turbine-driven AFW pump supplies both steam generators.

The motor-driven AFW pumps are protected against runout conditions by flow-limiting valves; the turbine-driven pump flow is limited by its governor so it will not achieve runout conditions. Because the main AFW system pumps are all located in the same room and subject to a high energy line break (HELB), a standby AFW system was installed independent of the main AFW system. The standby AFW system includes two motor-driven pumps (200 gpm each) and is manually initiated. Each standby AFW pump supplies one steam generator. The main AFW system complies with safety-grade and IEEE Std 279-1971 [8] requirements. The compliance of the remainder of the above systems with safety-grade and IEEE Std 279-1971 requirements was not reviewed.

The environmental qualification of safety-related electrical and mechanical components is being reviewed separately by the NRC and is not within the scope of this review.

The review did not determine whether the instrumentation that the operator relies upon to follow the accident and isolate the affected steam generator conforms with the criteria in ANS/ANSI-4.5-1980 [11] and Regulatory Guide 1.97 [12].

As part of the Systematic Evaluation Program (SEP), the NRC performed an evaluation of SEP topics VI-2.D, "Mass and Energy Release for Possible Pipe Break Inside Containment," and VI-3, "Containment Pressure and Heat Removal Capability," [4] for the Ginna plant. As part of this evaluation, the NRC performed an analysis to determine the containment pressure and temperature response. The analysis determined that the highest peak pressure and temperature were produced assuming hot standby conditions (zero power), failure of a containment spray pump, and AFW flow of 200 gpm. This worst-case accident produced a calculated peak containment pressure of 71.1 psig at 91 seconds. The containment design pressure is 60 psig; thus the analysis determined the containment design pressure was exceeded. In response to the draft version of the NRC evaluation of SEP topics VI-2.D and VI-3, RG&E

identified several conservatisms in the analysis for the MSLB. RG&E performed a reevaluation of the containment conditions following a MSLB by reconstructing the worst-case transient produced by the NRC analysis, evaluating the assumptions used, and then removing some of the conservatism to produce a more reasonable result. The RG&E reanalysis determined that for the worst case (hot standby and failure of containment spray pump), a peak containment pressure of 57.7 psig occurred at 128.6 seconds. The mass normally associated with auxiliary feedwater was not included in the analysis; the effect of auxiliary feedwater on peak containment pressure would be negligible since the mass added during the time frame of interest is a very small fracton of the secondary side inventory (less than 1%). From the Licensee's response [5], the NRC drew the following conclusion in Appendix B of the evaluation report on SEP topics VI-2.D and VI-3 [16]:

> "The calculated peak containment pressure for a MSLB accident is less than the containment design pressure."

3.1.3 Conclusion

The Licensee's responses [3, 5] and the NRC Evaluation Reports [4, 16] adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the design of the mitigating systems satisfy the NRC's acceptance criteria. Regarding Item 1, it is concluded that there is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. In addition, since the AFW pumps are protected from experiencing runout flow conditions, the pumps will be able to carry out their intended function without damage.

3.2 REVIEW OF REACTIVITY INCREASE ANALYSIS

The requirement from IE Bulletin 80-04, Item 2, is as follows:

> "Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return-to-power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water

sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return-to-power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient."

3.2.1 Summary of Licensee Statements and Conclusions

In regard to the reactivity increase resulting from a MSLB with continued feedwater addition, the Licensee stated [3]:

"Westinghouse Electric Corporation performed the original steam break analysis for Ginna as reported in the FSAR and a reanalysis submitted to the NRC in September 1975. Westinghouse has reviewed the assumptions made for main and auxiliary feedwater flow as they apply to licensing basis steam line break transients. Several of the relevant assumptions used in all core transient analyses follow, and are further explained in the Ginna FSAR.

1. The reactor is assumed initially to be at hot shutdown conditions, at the minimum allowable shutdown margin.

2. For the Condition IV breaks, i.e., double-ended rupture of a main steam pipe, full main feedwater is assumed from the beginning of the transient at a very conservative cold temperature.

3. All auxiliary feedwater pumps are initially assumed to be operating, in addition to the main feedwater. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure.

4. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, approximately 10 seconds after the

break occurs, while auxiliary feedwater is assumed to continue at its initial flow rate.

5. Main feedwater flow is completely terminated following feedwater isolation.

Based on the manner in which the analysis is performed for Westinghouse plants, the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater runout (or failure of runout protection where applicable) is minimal. Greater feedwater flow during the large steamline breaks serves to reduce secondary pressures, accelerating the automatic safeguards actions, i.e. steam line isolation, feedwater isolation and safety injection. The assumptions described above are therefore appropriate and conservative for the short-term aspect of the steamline break transient.

The auxiliary feedwater flow becomes a dominant factor in determining the duration and magnitude of the steam flow transient during later stages in the transient. However, the limiting portion of the transient occurs during the first minute, both due to higher steam flows inherently present early in the transient and due to the introduction of boron to the core via the safety injection system.

In conclusion, Westinghouse has evaluated the effect of runout auxiliary feedwater flows in the core transient for steamline break, and based on this evaluation, has determined that the assumptions presently made are appropriate for use as a licensing basis. The concerns outlined in the introduction to IE Bulletin 80-04 relative to, 1) limiting core conditions occuring during portions of the transient where auxiliary feedwater flow is a relevant contributor to plant cooldown; and 2) incomplete isolation of main feedwater flow, are not representative of the Westinghouse NSSS designs and associated Balance of Plant requirements.

The most limiting steam line break determined by Westinghouse was analyzed by Exxon Nuclear Co., Inc. and presented in XN-NF-77-40 Supplement 1, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant," March 1980. This transient occurs at hot zero power with outside power available and the break occurring at the exit of the steam generator. The Exxon analysis does not specifically account for auxiliary feedwater. However, the Steam Generator heat transfer model, using constant heat transfer coefficients, continues to calculate heat transfer from the primary to the secondary side after the broken steam generator has been estimated to be empty. If auxiliary flow was specifically accounted for, its effect would be negligible during the initial portion of the transient and would have minimal effect during later portions of the transient since by the time the broken steam generator empties, the total system reactivity is negative and core power is decreasing. The additional reactivity addition associated with the slight cooldown due to runout flow is more than negated by the boron reactivity inserted by safety injection. Therefore, the severity of the transient is not increased."

3.2.2 Evaluation

The Licensee's analysis of the core reactivity increase resulting from a MSLB with continued feedwater addition was reviewed in order to evaluate whether the following acceptance criteria were met:

- o Criterion 1.c - Ability to detect and isolate the damaged steam generator
- o Criterion 1.d - Changes in core reactivity increase
- o Criterion 3 - Analysis assumptions.

The Exxon Nuclear analysis [17] of the reactivity increase resulting from a MSLB and Reference 3 were reviewed. From that review, it was determined that the analysis is conservative in its assumptions and that the assumptions are in accordance with those in Acceptance Criterion 3, with the exception that runout AFW flow was not considered.

In the worst-case MSLB, which assumes no load conditions, a double-ended rupture at the steam generator exit, with offsite power available, a peak reactivity of about +0.004 occurs at 20 seconds which produces a maximum core power of 39% at 91.4 seconds. At this time, 20,000 ppm boron solution reaches the core, rapidly shutting down the reactor. The calculated return-to-power does not result in a violation of the specified acceptable fuel design limits.

The Licensee's conclusion that the core transient for the MSLB is insensitive to AFW flow is valid for the following reasons:

- o Early in the transient, the primary-to-secondary heat transfer rate (from the blowdown of the initial steam generator mass) is several orders of magnitude greater than that contributed by the additional AFW flow due to runout.

- Later in the transient (when the majority of the initial mass has blown down), AFW flow becomes a dominant factor in determining the magnitude and duration of the transient.
- The limiting core conditions will occur within the first minute due to the initial high cooldown rate contributing to the reactivity addition which is terminated by the introduction of 20,000 ppm boron solution into the core region.

Since the limiting core conditions occur before the AFW flow becomes a major contributing factor, it can be concluded that the core transient is insensitive to the contribution of AFW flow, and therefore the assumptions of the FSAR analysis remain valid.

3.2.3 Conclusion

The Licensee's response and reactivity response analysis [17] adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified, and although a reactor return-to-power is predicted, there is no violation of the specified acceptable fuel design limits. Therefore, the reactivity increase analysis [17] of the reactivity increase resulting from a MSLB remains valid.

3.3 REVIEW OF CORRECTIVE ACTIONS

The requirement from IE Bulletin 80-04, Item 3, is as follows:

"If the potential for containment overpressure exists or the reactor return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed."

3.3.1 Summary of Licensee Statements and Conclusions

The Licensee stated [3]:

"Since neither the potential for containment overpressurization nor the reactor-return-to-power response worsens no corrective action is required."

3.3.2 Evaluation and Conclusion

The Licensee's analysis determined that neither a containment over-pressurization nor a reactor return-to-power with a violation of the specified acceptable fuel design limits would occur from a MSLB. Therefore, it is concluded that no further action regarding IE Bulletin 80-04 is required of RG&E for the Ginna Nuclear Power Plant.

4. CONCLUSIONS

With respect to the R. E. Ginna Nuclear Power Plant, conclusions regarding Rochester Gas and Electric Corporation's response to IE Bulletin 80-04 are as follows:

- There is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition.
- The AFW pumps are protected from the effects of runout flow and therefore can be expected to carry out their intended function during the MSLB event.
- All potential water sources were identified and, although a reactor return-to-power is predicted, there is no violation of the specified acceptable fuel design limits. Therefore, the Exxon Nuclear [17] MSLB reactivity increase analysis remains valid.
- No further action regarding IE Bulletin 80-04 is required.

5. REFERENCES

1. "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition"
   NRC Office of Inspection and Enforcement, February 8, 1980
   IE Bulletin 80-04

2. "Overpressurization of the Containment of a PWR Plant after a Main Line Steam Break"
   NRC Office of Inspection and Enforcement, October 1, 1979
   IE Information Notice 79-24

3. L. D. White, Jr. (RG&E)
   Letter to B. H. Grier (NRC, Region I)
   Subject: Response to IE Bulletin 80-04
   April 30, 1980

4. D. M. Crutchfield (NRR, ORB #5)
   Letter to J. E. Maier (RG&E)
   Subject: "Systematic Evaluation Program (SEP) for the R. E. Ginna Nuclear Power Plant - Evaluation Report on Topics VI-2.D and VI-3"
   November 3, 1981

5. J. E. Maier (RG&E)
   Letter to D. M. Crutchfield (NRR, ORB #5)
   Subject: SEP Topics VI-2.D and VI-3
   February 1, 1982

6. Robert Emmett Ginna Nuclear Power Plant
   Final Facility Description and Safety Analysis Report, through Supplement 11
   Rochester Gas and Electric Corporation, March 1969

7. Technical Evaluation Report
   "PWR Main Steam Line Break with Continued Feedwater Addition - Review of Acceptance Criteria"
   Franklin Research Center, November 17, 1981
   TER-C5506-119

8. "Criteria for Protection Systems for Nuclear Power Generating Stations"
   Institute of Electrical and Electronics Engineers, New York, NY, 1971
   IEEE Std 279-1971

9. Standard Review Plan, Section 4.2
   "Fuel System Design"
   NRC, July 1981
   NUREG-0800

10. Standard Review Plan, Section 15.1.5
"Steam System Piping Failures Inside and Outside of Containment (PWR)"
NRC, July 1981
NUREG-0800

11. "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"
American Nuclear Society, Hinsdale, IL, December 1980
ANS/ANSI-4.5-1980

12. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"
Rev. 2
NRC, December 1980
Regulatory Guide 1.97

13. "Single Failure Criteria for PWR Fluid Systems"
American Nuclear Society, Hinsdale, IL, June 1976
ANS-51.7/N658-1976

14. "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
Rev. 3
NRC, February 1976
Regulatory Guide 1.26

15. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"
Rev. 1
NRC, July 1981
NUREG-0588

16. T. Speis (NRR, Assistant Director for Reactor Safety)
Memorandum to G. Lainas (NRR, Assistant Director for Safety Assessment)
Subject: Systematic Evaluation Program (SEP) for the R. E. Ginna Nuclear Power Plant - Evaluation Report on Topics VI-2.D and VI-3 (Docket No. 50-244)
March 31, 1982

17. "Plant Transient Analysis for the R. E. Ginna Unit I Nuclear Power Plant"
Exxon Nuclear Company, Inc.
XN-NF-77-40, Supplement 1
March 3, 1980