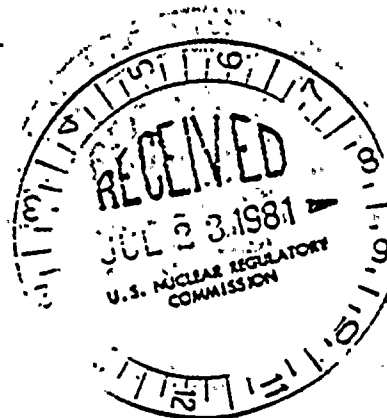


July 21, 1981

D  
Docket No. 50-244  
LS05-81-07-070



Mr. John E. Maier, Vice President  
Electric and Steam Production  
Rochester Gas & Electric Corporation  
89 East Avenue  
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPICS II-1.8, POPULATION DISTRIBUTION AND  
III-4.D, SITE PROXIMITY MISSILES - R. E. GINNA

Enclosed are the staff's final evaluations of SEP Topics II-1.8 and III-4.D for the R. E. Ginna Nuclear Power Plant. These evaluations are based on our review of your topic safety assessment reports submitted by letters dated April 15, 1981 and April 16, 1981, respectively.

You will note that we have revised your calculated population density which is more properly obtained by dividing the total population within a given distance by the total area of the complete circle (including both level and water) whose radius is the distance of interest.

This completes our evaluation of Topics II-1.8 and III-4.D.

These evaluations will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

*for* *Thomas V. Wambach*  
Dennis M. Crutchfield, Chief  
Operating Reactors Branch No. 5  
Division of Licensing

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PDR ADOCK 05000244  
PDR

Enclosure:

AT-23-NI



Mr. John E. Maier

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R. E. GINNA  
SYSTEMATIC EVALUATION PROGRAM TOPIC  
II-1.B, POPULATION DISTRIBUTION

I. INTRODUCTION

The safety objective of this topic is to ensure that the previously-established low population zone and population center distance specified for the site are compatible with the current population distribution, and are in accordance with the guidelines of 10 CFR Part 100.

II. REVIEW CRITERIA

Sections 100.10 and 100.11 of 10 CFR Part 100, "Reactor Site Criteria" provides the site evaluation factors which should be considered when evaluating sites for nuclear power reactors. These sections include guidelines for determining the exclusion area, low population zone and population center distance.

III. RELATED SAFETY TOPICS

Topic II-1.A, reviews the licensee's control over the exclusion area. Various other topics will evaluate the capability of the plant to meet the dose criteria of 10 CFR Part 100 at the exclusion area boundary and low population zone. The adequacy of emergency preparedness planning for the area surrounding the plant including the low population zone is being assessed by the Commission in a separate review effort.

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IV. REVIEW GUIDELINES

The review has been conducted in accordance with Standard Review Plan (SRP) Section 2.1.3, "Population Distribution."

V. EVALUATION

The R. E. Ginna site is in the township of Ontario, in the northwest corner of Wayne County, New York, on the north shore of Lake Ontario about 20 miles ENE of the center of the City of Rochester and 40 miles WSW of Oswego. The land surrounding the site is primarily of an agrarian nature and sparsely populated. There are no substantial population centers, industrial complexes, transportation arterials, parks, or other recreational facilities within a three mile radius of the Ginna site.<sup>2</sup> The City of Rochester is the largest population center within a 50 mile radius of the site (241,539 people, with 701,745 in the metropolitan area<sup>7</sup>). The nearest community with a population of 1,000 or more is the Town of Ontario with its center located about 3½ miles from the site. The preliminary estimated 1980 census for the Town of Ontario is 7,452.<sup>7</sup>

To develop the Wayne County and Monroe County Radiological Emergency Response Plans for the R. E. Ginna Nuclear Power Station, a recent survey of the population within a five-mile radius was completed. Figure J-2 from the Wayne County Radiological Response Plan, reproduced as Figure 1

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of this evaluation, details the population within 5 miles of Ginna, based on preliminary 1980 population estimate. RG&E estimates that 10,864 persons reside within five miles of the plant, a density of 138 persons per square mile averaged over the entire area. (It should be noted that this figure compares favorably with the 1980 population projection of 10,934 persons shown in Figure 2.4-2 of the Ginna FSAR, which was published in 1968).

Other than the residents of the area, there are no large groups of transients within five miles of the site. The only parks near the site are Webster Beach Park in Monroe County, approximately 6 miles west of the plant site, and B. Forman Park in Wayne County, approximately 8 miles east of the plant site. There are no federal recreational facilities in the area. There are no state parks, public campsites, or special use areas within ten miles of the plant.<sup>2</sup> Wayne County does have a migrant labor population, primarily for apple picking, during the June-October season. Approximately 115 farmworker camps of five or more persons are scattered throughout Wayne County<sup>8</sup>, with a total population of about 4400 migrants. Information from Rural New York Farmworker Opportunities shows that there are only 12 camps, with about 130 migrants, located in the vicinity of the Ginna site.<sup>10</sup>

The nearest population center to the Ginna site containing more than 25,000 residents is the "Rochester urbanized area," whose eastern boundary is about ten miles from the site.<sup>2</sup> The only other population center of more than 25,000 persons is the City of Auburn (population 32,442),<sup>7</sup> located more than 40 miles SE of the site.

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The low population zone specified for the Ginna site is the area within a 3 mile (4,827 meter) radius of the plant.<sup>9</sup> A review of current population estimates and projected growth estimates indicate that the population growth in the area since the plant received an operating license in 1969 has been modest, and this trend is expected to continue. No population center of 25,000 residents has developed, or appears likely to develop, closer than the eastern boundary of the Rochester urbanized area.

VI. CONCLUSION

The staff concludes that the low population zone and population center distances specified for the Ginna site is in conformance with the requirements of 10 CFR Part 100 in that the population center distance is more than one and one-third times the distance from the reactor to the outer boundary of the low population zone (10 miles vs. 3 miles).

We further conclude that the site conforms to the current licensing criteria. This completes the evaluation of SEP Topic II-1.B for the Ginna site.

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VII. REFERENCES

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.4.
2. Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Section 2.1.3, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Wayne County Radiological Emergency Response Plan, Draft Rev. B, November 1980.
6. Monroe County Radiological Emergency Response Plan Draft, Rev. B, November 1980.
7. Preliminary Report, 1980 Census of Population and Housing, New York, published by the Bureau of the Census, U. S. Department of Commerce, February 1981.
8. Conversation with the New York State Health Department, April 13, 1981.
9. Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission in the Matter of Rochester Gas and Electric Corporation Robert Emmett Ginna Nuclear Power Plant Unit No. 1, Docket No. 50-244 (SER), Section 2.1, June 19, 1969.
10. Letter, Thomas J. Harris, RNYFO, to George Wrobel, RG&E, April 10, 1981.

11. Rochester Gas and Electric Corporation, Ginna Nuclear Station Radiation Emergency Plan, Proposed January 1981.
12. New York State Radiological Emergency Preparedness Plan, December 1980.

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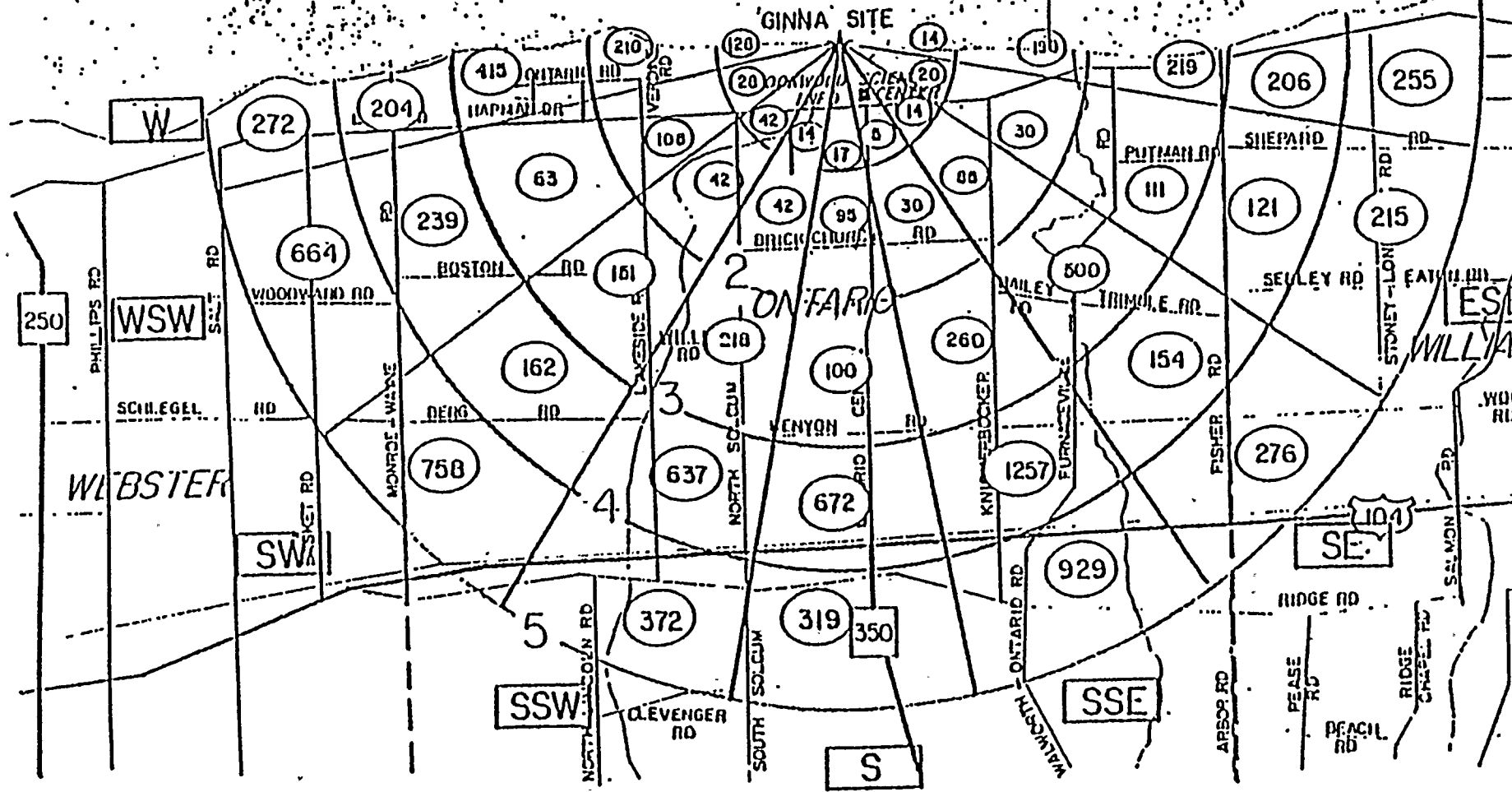
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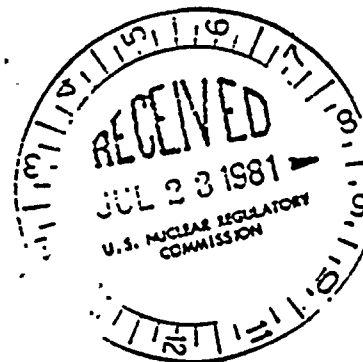
(xxx) 1980 POPULATION ESTIMATES

FIGURE 1  
0-5 MILE 1980 POPULATION ESTIMATE



July 21, 1981

Docket No. 50-244  
LS05-81-07-070



Mr. John E. Maier, Vice President  
Electric and Steam Production  
Rochester Gas & Electric Corporation  
89 East Avenue  
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPICS II-1.8, POPULATION DISTRIBUTION AND  
III-4.D, SITE PROXIMITY MISSILES - R. E. GINNA

Enclosed are the staff's final evaluations of SEP Topics II-1.8 and III-4.D for the R. E. Ginna Nuclear Power Plant. These evaluations are based on our review of your topic safety assessment reports submitted by letters dated April 15, 1981 and April 16, 1981, respectively.

You will note that we have revised your calculated population density which is more properly obtained by dividing the total population within a given distance by the total area of the complete circle (including both level and water) whose radius is the distance of interest.

This completes our evaluation of Topics II-1.8 and III-4.D.

These evaluations will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

*for Thomas V. Wambach*  
Dennis M. Crutchfield, Chief  
Operating Reactors Branch No. 5  
Division of Licensing

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PDR

Enclosure:

ATTN: 01



R. E. GINNA  
SYSTEMATIC EVALUATION PROGRAM TOPIC  
TOPIC III-4.D, SITE PROXIMITY MISSILES  
(INCLUDING AIRCRAFT)

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for a site proximity missile.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis." of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic II-1.C, "Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial and Military Facilities" provides a description of the potential missile hazards.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.3, "Evaluation of Potential Accidents," 3.5.1.5, "Site Proximity Missiles (except Aircraft)," and 3.5.1.6, "Aircraft Hazards."



V. EVALUATION

The potential for hazardous activities in the vicinity of the Ginna plant has been addressed in SEP topic II-1.C, "Potential Hazards due to Industrial, Transportation, Institutional and Military Facilities". As indicated therein, there is little industrial activity near the plant. The distances to the nearest land transportation routes are such (about 1700 feet to the nearest highway, and 3 1/2 miles to the nearest railroad) that the risk associated with potential missiles from transportation accidents on these routes are within the SRP 2.2.3 guidelines. Similarly, the nearest large gas pipelines are about six miles from the plant, and do not pose a missile threat to the plant. Major Lake Ontario shipping routes are also sufficiently far away (about 23 miles) so as not to present a credible missile hazard from lake traffic. There are no military facilities or activities near the plant which would create a missile hazard.

The review of SEP Topic II-1.C also evaluated the potential for aircraft becoming a missile hazard, both in connection with the operation of the Williamson Flying Club Airport, which is about ten miles ESE of the plant, and due to commercial air traffic in and out of Rochester via federal airways V2N and V2, which are 2 1/2 and 10 miles from the plant site.

As evaluated in Topic II-1.C, it was determined that, since the Williamson Flying Club Airport expected a maximum of only 5000 operations per year, and is about 10 miles from the site, the criteria in III.3.a and III.3.b of SRP 3.5.1.6 were met, and there is no need to determine the probability of an aircraft crash into the plant. Further, the hazard to the plant from commercial aircraft use of airways V2 and V2N was shown to be only  $5.1 \times 10^{-8}$  and  $1.4 \times 10^{-8}$  per year, respectively. No danger to the plant from commercial airline traffic is thus expected.

Conclusion

Since current regulatory criteria are met with respect to SEP Topic III-4.D, "Site Proximity Missiles", it can be concluded that this topic is complete for the R. E. Ginna site. No additional review for this topic is required during the SEP integrated assessment.

VI. REFERENCES

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.5.
2. Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Sections 2.2.1, 2.2.2, 2.2.3, and 3.5.1.6, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Sterling Power Project Nuclear Unit No. 1, Preliminary Safety Analysis Report Addendum, Rochester Gas and Electric, Volume 1, Sections-2.1 and 2.2.
6. U.S. Nuclear Regulatory Commission Regulatory Guide 1.91, Rev. 1, February 1978.
7. Letter, John E. Maier, RG&E, to Dennis M. Crutchfield, NRC, SEP Topic II-1.C, "Potential Hazards Due to Transportation, Industrial, Institutional and Military Facilities", April 15, 1981.







UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555  
February 22, 1982

*McKenina*

Docket No: 50-244  
LS05-82-02-091

Mr. John E. Maier  
Vice President  
Electric and Steam Production  
Rochester Gas & Electric Corp.  
89 East Avenue  
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: GINNA - SEP TOPIC III-5.A, EFFECTS OF PIPE BREAK ON SYSTEMS  
STRUCTURES AND COMPONENTS INSIDE CONTAINMENT

By letter dated June 30, 1981, the staff issued a draft safety evaluation on SEP Topic III-5.A which identified ten open items for further consideration.

Your letter of October 1, 1981, provided responses to the above items. Based on our review of these letters, we conclude that although several of the items have been resolved, additional information is needed to close out the remaining open items.

Enclosure 1 discusses each of the open items and their status. Enclosure 2 summarizes the information that you are requested to provide. Enclosure 3 is the revised safety evaluation report for Topic III-5.A, including staff guidelines for resolution of high energy pipe break locations where remedial modifications are impractical. This safety evaluation will be a basic input to the integrated plant safety assessment for your facility. Resolution of the open items will be addressed in the integrated assessment.

You are requested to provide your schedule for completion of the items identified in Enclosure 2 within 30 days of receipt of this letter.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

*Dennis M. Crutchfield*  
Dennis M. Crutchfield, Chief  
Operating Reactors Branch No. 5  
Division of Licensing

Enclosures: As stated

cc w/enclosures: See next page



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STATUS OF OPEN ITEMS FROM DRAFT SER

- References: (1) Letter, D. Crutchfield (NRC) to J. Maier (RG&E), dated June 30, 1981
- (2) Letter, J. Maier (RG&E) to D. Crutchfield (NRC), dated October 1, 1981.

The open items identified by the staff and the licensee responses are provided in References 1 and 2 respectively. The present status of each item, keyed to the numbering in the references, is given below.

1. The first open item was concerned with the general assumptions made in this analysis. One of the basic assumptions of this topic assessment was that a check valve in an incoming line would prevent primary system blowdown in the event of a pipe break upstream of the valve. This is true provided the check valve closes. Adequate assurance must be demonstrated that these normally open check valves will fulfill their assumed isolation function.
2. On a mechanistic basis, the postulated break locations in the main steam line would not impact the containment wall. For the feedwater lines, the licensee provided an analysis of the structural integrity of the containment. As a bounding analysis, the steam line break thrust force was used. The results show that the containment remains intact, even neglecting the containment liner plate. This issue is considered to be resolved.
3. The licensee has provided the piping stress results for the "B" steam line. None of the locations exceeded the stress criteria of  $0.8 (1.2S_h + S_A)$ . Accordingly, breaks were postulated at the terminal ends and at the two highest-stressed intermediate locations. None of these breaks would cause the crane to fall. Therefore, this item is resolved.
4. For the "A" accumulator line a mechanistic evaluation was performed. The stresses in this line were all below the criteria, so breaks were postulated at terminal ends and at the two intermediate locations of highest stress. One of these points was inside the loop compartment where no adverse interactions would occur.

The second point is located just on the reactor side of the (normally locked open) motor-operated valve. At this location no adverse pipe whip interactions will occur. Adequate protection from jet impingement effects must be provided. If remedial measures to provide this protection can be shown to be impractical, fracture mechanics evaluations can be performed to establish that conditions that could lead to a double-ended rupture do not exist as discussed in the guidance provided in the Attachment to Enclosure 3. The effect of a break in the two inch accumulator level taps on nearby instrument circuits is still under review by the licensee.

5. For the pressurizer surge line, since some jets could affect safety-related equipment, analyses similar to those described in item 4 above should be provided.
6. A mechanistic evaluation of the pressurizer spray line was performed. Since the calculated stresses did not exceed the criteria, breaks were postulated at the terminal ends and at the two highest-stressed locations. None of these break locations would prevent operation of the sump valves and therefore, this item is resolved.
7. For the letdown line, licensee evaluation of the effects on cables and cable trays is continuing. Adequate protection for instrumentation should be provided.
8. The situation for the steam generator blowdown lines is similar to item 7 for the instrumentation. With respect to the fan coolers, this size break is not limiting with respect to containment pressure/temperature reduction capability. The containment spray system would be available for containment cooling. As for item 7 above, final resolution will occur after the effects on the cable trays are evaluated.
9. The licensee has provided the requested references to the subcompartment analyses performed for Reactor Coolant System (RCS) guillotine breaks. In addition, the analysis discussed above in item 2 showed that a 30 inch steam line would not penetrate a 30 inch concrete wall. Since the pipes under consideration in the compartment are 10 inch diameter lines, we consider that this concern is resolved.
10. Pipe breaks were not postulated in the primary loop on the basis of the work done under TAP A-2. We concur with this approach. However, the SEP branch intends to evaluate the effects on safety-related equipment of the jet loads resulting from the crack sizes associated with these analyses.

Enclosure 2

REQUEST FOR ADDITIONAL INFORMATION

1. Please provide your basis for assurance that check valves relied upon to prevent primary system blowdown will fulfill their function.
2. The following breaks are still under evaluation by you:
  - (a) accumulator level taps
  - (b) letdown
  - (c) steam generator blowdown

Please provide your proposed schedule for these further evaluations. Adequate protection for instrumentation circuits should be provided.

3. For the 10 inch accumulator line and the 10 inch pressurizer surge line, provide your planned resolution for possible jet impingement interactions. If remedial modifications are impractical, the guidance in the attachment to Enclosure 3 may be used to provide reasonable assurance that mitigation of pipe break effects for these lines is unnecessary.

SEP TOPIC III-5.A  
EFFECTS OF PIPE BREAK ON STRUCTURES, SYSTEMS AND  
AND COMPONENTS INSIDE CONTAINMENT

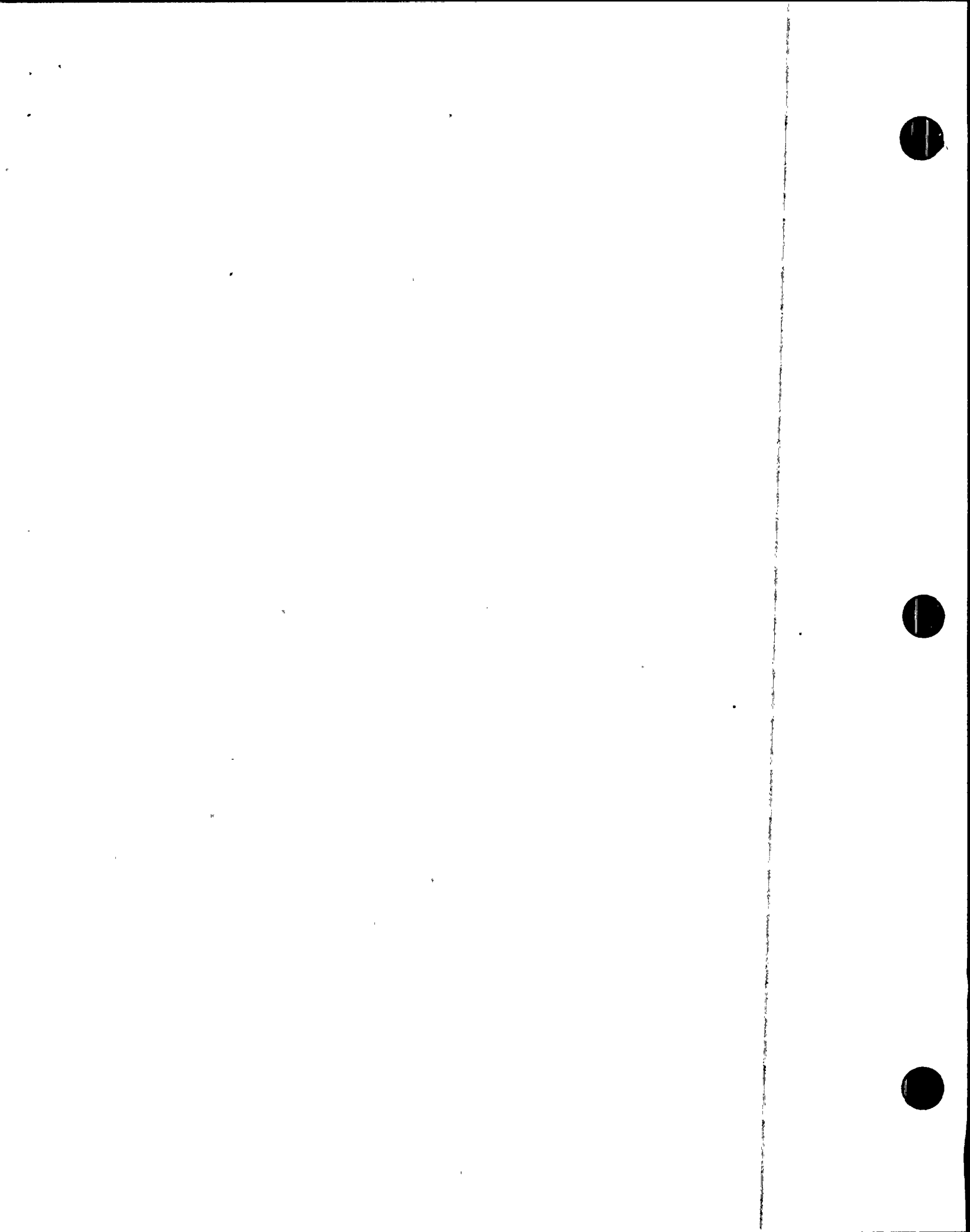
R. E. GINNA

(FEBRUARY 1982)



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- I. INTRODUCTION
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  - C. SAFETY RELATED EQUIPMENT
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  - B. INTERACTION STUDIES
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## I. INTRODUCTION

The safety objective of Systematic Evaluation Program (SEP) Topic III-5.A, "Effects of Pipe Break on Structures, Systems and Components Inside Containment," is to assure that pipe breaks would not cause the loss of needed function of "safety-related" systems, structures and components and to assure that the plant can be safely shutdown in the event of such breaks. The needed functions of "safety-related" systems are those functions required to mitigate the effects of the pipe break and safely shutdown the reactor plant.

## II. REVIEW CRITERIA

The current criteria for review of pipe breaks inside containment are contained in Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," including its attached Branch Technical Position, Mechanical Engineering Branch 3-1 (BTP MEB 3-1).

## III. RELATED SAFETY TOPICS AND INTERFACES

1. This review complements that of SEP Topic VII-3, "Systems Required for Safe Shutdown."
2. The environmental effects of pressure, temperature, humidity and flooding due to postulated pipe breaks are evaluated under Unresolved Safety Issues (USI) A-24, "Qualification of Class 1E Safety-Related Equipment."
3. The effects of potential missiles generated by fluid system ruptures and rotating machinery are evaluated under SEP Topic III-4.C, "Internally Generated Missiles."
4. The effects of containment pressurization are addressed under SEP Topic VI-2.D, "Mass and Energy Release for Possible Pipe Break Inside Containment."
5. The original plant design criteria in the areas of seismic input and analysis design criteria are evaluated under SEP Topic III-6, "Seismic Design Consideration."

## IV. REVIEW GUIDELINES

On September 7, 1978, the SEP Branch sent a letter (Reference 1) to Rochester Gas & Electric Corporation (RG&E) requesting an analysis of the effects of postulated pipe breaks on structure, systems and components inside containment. In that letter, the staff included a position that stated three approaches were appropriate for postulating breaks in high energy piping systems ( $P=275$  psig or greater or  $T=200^{\circ}\text{F}$  or greater). The approaches are:

1. Mechanistic
2. Simplified Mechanistic
3. Effects Oriented

The staff further stated that combinations of the three approaches could be utilized if justified. The details of those three approaches are described in Reference 1.

V. DISCUSSION

A. Background

In a letter dated February 9, 1979 (Reference 2), RG&E submitted a list of high energy lines inside containment. Representatives of the NRC and RG&E staff met at the Ginna site on March 13 and 14, 1979, to discuss the analyses done by the licensee on this topic. As a result of this meeting, the licensee submitted on September 12, 1979 a report (Reference 3) on the effects of breaks in these lines on safety-related equipment. This review utilized the effects-oriented approach for the high energy line breaks analyzed. In this approach, breaks were postulated at any location along the line, and were chosen to produce the greatest jet impingement or pipe whip loadings on essential equipment. Also, the assumed plane of motion was that which produced the most adverse effects unless otherwise justified.

B. Analysis Assumptions

The following assumptions were made by the licensee:

1. High energy fluid systems are systems with operating temperature greater than 200°F or operating pressure greater than 275 psig. In accordance with Branch Technical Position (BTP) MEB 3-1, breaks are not postulated in piping of systems that qualify as high energy systems for only short operational periods (i.e., less than 2% of the time the system operates as a moderate energy system). Pipes less than one inch (1") in diameter were also eliminated in accordance with Regulatory Guide 1.46.
2. Pipe of a given section modulus will not cause a loss of function in pipe of equal or larger section modulus as a result of pipe whip or jet impingement.
3. Pipe whip can only occur in the section of pipe which is attached to a sustained high energy source. Credit is taken for all closed or automatically closed valves (e.g., check valves) in the piping section that could terminate flow.

4. The jet impingement force (calculated to be less than 200 pounds) due to breaks in the 2" diameter lines fed by the positive displacement charging pumps will not impair functioning of equipment.
5. In addition to the equipment affected by the break, a single independent failure of an active component inside containment is considered.

C. Safety-Related Equipment

Safety-related equipment includes systems needed to mitigate the effects of the line breaks and to bring the reactor to safe shutdown.

Breaks inside containment generally result in or have the same effect as loss of coolant accidents or steam/feed line breaks. Engineered safety features are required to mitigate these breaks.

Other breaks (such as accumulator line breaks) do not result in a loss of inventory or energy from the reactor coolant system and thus require only normal safe shutdown systems such as CVCS.

Systems that are all or partially inside containment are:

- Safety Injection (SI) - two trains one to each cold leg, no active components inside containment
- Low Pressure Safety Injection (LPSI) - two trains which pump water to the injection nozzles on the vessel through motor-operated valves (MOVS) 852A and B, which must change position on receipt of a safety injection signal
- Accumulators - directed to each cold leg, no active components
- Containment Spray - two trains to spray headers in containment, no active components inside containment
- Containment Fan Coolers and Service Water - four fan coolers which must operate to provide cooling; service water has no active components inside containment
- Sump Recirculation - two lines from the sump to Emergency Core Cooling System (ECCS) pumps, no active components inside containment
- Residual Heat Removal - one drop line, one return line, each with two MOVs

- Chemical and Volume Control (CVCS) Charging and Letdown - two physically separated charging paths
- Standby Auxiliary Feedwater System - ties into main feedwater lines, no active components inside containment; auxiliary feedwater system is totally outside containment
- Essential Instrumentation - pressurizer pressure, steam generator level.

## VI. EVALUATION

### A. Assumptions and Criteria

As discussed earlier, lines separated from an energy reservoir by a check valve were not assumed to have sufficient energy to whip or produce jets. For long runs of large piping, the energy stored within the pipe volume from the break to the valve could be sufficient to form a jet. For Ginna, however, the only pipes for which the check valve separation is utilized to limit interactions are 2" pipes, so this effect is not expected to be significant.

However, assurance must be provided that the normally open check valve closes sufficiently so that the dynamic forces from the reactor side are not significant.

The staff concurs that use of the pipe section modulus is an appropriate measure of relative strength of pipes. The licensee has assumed that a pipe of larger section modulus will break a pipe of smaller section modulus, but a smaller section modulus was not considered to affect a larger section modulus. In accordance with staff positions transmitted on January 4, 1980 (Reference 4), the effects of jet impingement loads should be considered and evaluated regardless of the magnitudes of the section modulus of impinged and postulated broken pipes. Therefore, the licensee should perform additional evaluations of the effects of jet impingement on equipment and piping. An acceptable jet model is described in Standard Review Plan Section 3.6.2.

Single failures of active components inside containment, such as the fan coolers or LPSI valves were considered by the licensee. Loss of offsite power was not specifically addressed in this study, but the staff has included consideration of the consequences in its review. In the safety injection and accumulator systems, a loss of offsite power, a single failure and a broken injection line would not prevent injection flow into the other loop. For the con-

tainment spray system, a loss of offsite power, single failure of a diesel and a rupture of one spray line could reduce containment heat removal capability below the minimum assumed in the LOCA analysis. Thus, a break that could affect a containment spray line should be further considered to see if the remaining systems are adequate.

#### B. Interaction Studies

For each of the postulated break locations, the licensee evaluated the effects on the essential equipment. In addition, the effects on other impacted equipment were considered to ensure that failure of such equipment would not exacerbate the break effects.

Several of the breaks would be confined within one of the loop compartments, would not affect the other train of safety injection, or the low pressure safety injection system, and therefore, would not prevent safe shutdown. Most breaks in loop compartments do not affect safety-related electrical equipment since this equipment is not located inside the compartments.

A high energy line is assumed to break an impacted line of smaller section modulus. If this impacted line is also a high energy line, the potential dynamic effects of that break must be concurrently considered. The check and isolation valves located close to the reactor connection on most of the high energy lines assure that, even if a line is broken by the initiating pipe break, there is insufficient energy to produce other effects from the second break. Such a situation arises with the accumulator line (from tank skirt to loop compartment wall). Breaks in this line can affect the Residual Heat Removal (RHR) outlet line. This line is itself a high energy line within the loop compartment, but is not a high energy line outside the compartment due to the two normally-closed isolation valves.

Within loop compartments or within the pressurizer compartment, the potential exists for high energy lines to impact other high energy lines, such as the RHR in line impacting a charging line. However, in general, the minimum engineered safety features (ESF) needed to mitigate these breaks are physically separated from the break and are thus unaffected.





Breaks in the primary loop of reactor coolant system (RCS) were not addressed by the licensee in this study on the basis of the work performed for USI A-2 (Asymmetric Blowdown Loads on Reactor Primary Coolant System).

Based on the interaction studies under the effects-oriented approach, several locations were identified with potentially unacceptable consequences for which further evaluation was necessary. The staff issued a draft safety evaluation on this topic on June 30, 1981 (Reference 5). The potentially unacceptable break locations were identified for further review.

The licensee responded to our draft evaluation on October 1, 1981 (Reference 6). For some high energy lines that could not be shown to be acceptable on an effects-oriented basis, a mechanistic evaluation was performed (see Reference 1). In this approach, stress analyses were performed to locate the most highly stressed points, which are the locations most likely to fail. Breaks must be postulated at all intermediate locations where the stress for the limiting normal and upset conditions exceeds 0.8 (1.2  $S_H + S_A$ ) and at terminal ends. If all stresses are below this criteria, at least two intermediate points, the highest two stresses, must be postulated.

Breaks at the highest stress locations did not result in unacceptable consequences for the main steam line and the pressurizer spray line. The effect on the containment wall of whip of the feedwater line was determined to be acceptable based on a bounding analysis of steam line impact. The results of this analysis showed that the penetration of the concrete is less than 14 inches, even neglecting the steel containment liner. Thus, a feedwater line break will not result in loss of structural integrity of the containment, and break consequences are considered to be acceptable.

For three lines, the licensee is continuing his review of jet effects on instrumentation and cable trays:

- (a) letdown line
- (b) steam generator blowdown line
- (c) accumulator level taps.

Although no adverse pipe whip interactions can occur, safety-related equipment must be adequately protected from jets resulting from failures of the accumulator line and the pressurizer surge line.

Pipe ruptures in the primary coolant loop were not postulated because of the A-2 "leak-before-break" technique, however, the staff will assess the effects of jets from crack sizes determined by that work.

## VII. CONCLUSIONS

The staff has reviewed the layout drawings, analyses and other information provided by the licensee. In addition, the staff toured representative locations on June 1-2, 1981 in the Ginna containment to observe the pipe configurations and proximity to safety related equipment. Based on these reviews, we conclude that the licensee has satisfactorily addressed the pipe whip and jet effects of high energy line breaks inside containment and has demonstrated an adequate level of protection subject to resolution of the following:

1. A basis must be provided for assuming that the normally open check valves relied upon for prevention of reactor blowdown will close.
2. The evaluation of effects on instrumentation circuits from breaks in letdown piping, steam generator blowdown piping and the accumulator level taps is still ongoing. An adequate level of protection for the instrumentation must be demonstrated.
3. For the accumulator line and the pressurizer surge line, adequate protection of safety-related targets must be provided. If remedial modifications are shown to be impractical, fracture mechanics evaluations may be performed. Guidelines for this analysis are provided in the attachment.
4. As discussed above, the staff will evaluate jets from cracks in the primary coolant loop.

## VIII. REFERENCES

1. Letter from D. Eisenhut (NRC) to L. D. White (RG&E), dated September 7, 1978.
2. Letter from L. D. White (RG&E) to D. Ziemann (NRC), dated February 9, 1979.
3. Letter from L. D. White (RG&E) to D. Ziemann (NRC), dated September 12, 1979.
4. Letter from D. Ziemann (NRC) to L. D. White (RG&E), dated January 4, 1980.
5. Letter from D. Crutchfield (NRC) to J. Maier (RG&E), dated June 30, 1981.
6. Letter from J. Maier (RG&E) to D. Crutchfield (NRC), dated October 1, 1981.

GUIDANCE FOR RESOLUTION OF HIGH  
ENERGY PIPE BREAK LOCATIONS  
WHERE REMEDIAL MODIFICATIONS  
ARE IMPRACTICAL

From the results of reviews conducted to date, the staff has concluded that the relocation of equipment or other modifications to mitigate the consequences of some postulated pipe breaks may be impractical due to physical plant configurations or other considerations. Therefore, the staff has determined that for specific locations where relocation of equipment or other modifications to mitigate consequences of pipe breaks are shown to be impractical, fracture mechanics evaluation of the piping should be performed to determine if unstable ruptures could occur in piping that contained service induced large undetected flaws.

The intent of the guidance provided by the staff is to provide reasonable assurance that the mitigation of pipe breaks are addressed. The approach taken is to provide assessment that condition which could lead to a double ended pipe rupture do not exist thereby making it unnecessary for high energy pipe break considerations to mitigate effects of a guillotine rupture. This would be accomplished using a defense in depth approach that is a combination of augmented inservice inspection (ISI), local leak detection and fracture mechanics evaluations. Augmented inservice inspections would be performed with the goal of detecting and limiting any service induced flaws to limits prescribed by the ASME B&PV Code, Section XI, approximately 10% thru wall. Should the flaws go undetected, a local leak detection system would be provided with the requisite sensitivity to identify leakage from a through crack, either longitudinal or circumferential, of a length of twice the wall thickness for minimum flow rates associated with normal (Level A) operating conditions. Fracture mechanics evaluations would be performed to determine that for a circumferential or longitudinal through crack of four wall thickness subjected to maximum ASME design code loads (Level D) that:

- (1) substantial crack growth does not occur.
- (2) local or general plastic collapse (instability) does not occur.
- (3) flow through the crack or the effects of a jet from the crack does not impair safe system shutdown.

To provide assurance that a double ended rupture could not occur by unanticipated loads being applied to a large undetected crack, a fracture mechanics evaluation would be performed to demonstrate that a through crack of a length of four times the wall thickness, 90° total circumferential length, or a larger crack if justified for system service experience would remain stable for local

fully plastic large deformation bending conditions. The basis for performance of this more conservative fracture mechanics evaluation to assure a double ended pipe rupture would not occur is as follows:

- (1) operating experience has shown that unanticipated and undefined loads in excess of design can and do occur in piping systems, i.e., water hammer events have failed piping system supports.
- (2) uncertainty in: (a) current analysis methods to accurately predict piping loads analysis and (b) prediction of the energy and frequency content of earthquakes and their effect on piping loads.
- (3) SEP criteria for evaluation of structures and system resistance to postulated earthquake loads depend on global structural ductility. This assumption is based on the ability to have load redistributions occur. For unflawed piping, the necessary local ductility is certainly provided. However, for flawed sections of piping the ability to sustain fully plastic behavior without crack instability is required to assure prudently that local ductility is preserved.

The details of the guidance for the combined augmented ISI, leak detection and fracture mechanics evaluations are attached as Enclosure 1.

ALTERNATIVE SAFETY ASSESSMENT FOR SELECTED  
HIGH ENERGY PIPE BREAK LOCATIONS  
AT SEP FACILITIES

This assessment is required only if a LWR high energy piping system (i.e., 275 psi, or higher; or 200 F or higher, etc.) is being considered. It is only required, if a postulated double ended pipe break would impair safe system shutdown by pipe whip (lacking pipe whip constraints) consequences, or by the consequences of the implied leakage or its jet action. The following guidance is for a safety assessment that may be permitted as an alternative to other system modifications or alterations for locations where the mitigation of the consequences of high energy pipe break (or leakage) have been shown to be impractical.

Guidance for Alternate Safety Assessment

The suggested guidance are as follows:

A. Detectability Requirements

Provide a leak detection system to detect through-cracks of a length of twice the wall thickness for minimum flow rates associated with normal (Level A) ASME B&PV Code operating conditions. Both circumferential and longitudinal cracks must be considered for all critical break or leak locations. Methods for estimation of crack opening areas are attached. Surface roughness of the crack should be considered.

B. Integrity Requirements

(1) Loads for Which Level D is Specified

- (a) Show that circumferential or longitudinal through-cracks of four wall thicknesses in length subjected to maximum Level D loading conditions do not exhibit substantial monotonic loading crack growth (e.g., staying below  $J_{IC}$  or  $K_{IC}$  by plastic zone corrected linear elastic fracture mechanics methods or a suitable alternative). Also assure that local or general plastic instability does not occur for these loading conditions and crack sizes.

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For  $4t$  flaws that are calculated to be greater than  $K_{IC}$  or  $J_{IC}$ , consideration will be given to; (1) flaw growth arguments, (2) postulation of small flaws sizes than  $4t$  if justified by leak detection sensitivity.

- (b) Under conditions in "B.(1)" show that the flow through the crack and the action of the jet through the crack will not impair safe shutdown of the system.

Acceptable methodology for the estimation of crack opening area for a circumferential through crack in a pipe in tension and bending and for longitudinal cracks subject to internal pressure are attached.

(2) Extreme Conditions to Preclude a Double-Ended Pipe Break

Using elastic-plastic fracture-mechanics or suitable alternative show that circumferential through-cracks will remain stable for local fully plastic large-deformation bending conditions under the following additional conditions:

- (a) Fully plastic bending of the cracked section is to be assumed, unless other load limiting local conditions (such as elbow collapse) dictate maximum bending loads, for all critical locations.
- (b) Assume that all system anchors are effective, but that other supports (such as hangers and snubbers) are inoperative unless especially justified.
- (c) Other as built displacement limits or constraints may be assumed as especially justified (such as displacement limits of a pipe running through a hole in a sufficiently strong concrete wall or floor, etc.).
- (d) Assume a through-crack size of  $4t$  or  $90^\circ$  total circumferential length whichever is greater; or a larger crack only if especially justified.
- (e) Assume large deformations means deformations proceeding to as built displacement limits or other especially justified limits.

(3) Material Properties

Conservative material properties should be used in the analyses. Sufficient justification must be provided for the properties, both weldment and base metal, used in the analyses.



C. Subcritical Crack Development

Consideration should be given to the types of subcritical cracks which may be developed at all locations associated with this type of analysis. From prior experience and/or direct analysis it should be shown that:

- (1) there is a positive tendency to develop through-wall cracks.
- (2) if there is a tendency to develop long surface cracks in addition to through-wall cracks, then it should be further demonstrated that the long surface crack will remain sufficiently shallow.

D. Augmented Inservice Inspection

Piping system locations for which corrective measures are not practicable should be inspected volumetrically in accordance with ASME Code, Section XI for a Class 1 system regardless of actual system classification.

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ESTIMATION OF STRESS INTENSITY FACTORS AND THE CRACK OPENING  
AREA OF A CIRCUMFERENTIAL AND A LONGITUDINAL  
THROUGH-CRACK IN A PIPE

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Introduction

Formulas for estimating the crack opening area are developed for a circumferential and a longitudinal through-crack in a pipe subjected to several types of loading. For the circumferential crack, estimation formulas are presented for axial force and bending moment applied to the pipe far from the cracked section and for internal pressure loading. For the longitudinal crack, an estimation formula for the case of internal pressure is presented.

Estimation is based on the method of linear elastic fracture mechanics, which requires the knowledge of the solution of stress intensity factor,  $K$ , for each problem. For the internal pressure loading,  $K$ -solutions are readily available for both circumferential and longitudinal cracks as functions of a single geometric parameter,  $\lambda (= a/\sqrt{Rt})$ , relating crack size and pipe geometry. Consequently, the crack opening area formulas are also formulated as functions of this single parameter. For the case of tension and bending of circumferential crack, however, the stress intensity factors are not formulated as functions of a single parameter and no simple formula is readily available. Therefore, in this discussion, a typical value of

mean radius to thickness ratio,  $R/t = 10$ , is specifically selected and formulation is made for this value. Estimation formulas are expected to yield a slight overestimate for  $R/t = 10$ . For smaller  $R/t$  ratios, degree of overestimate would increase. The formulas presented here may be used with a reasonable accuracy when  $R/t$  ratio is about 10. Formulas for the crack opening for these cases are not available in simple closed forms, but here moderately long power series approximations based directly on the estimating formulas for  $K$  are given.

#### A Circumferential Through-Crack in Tension and Bending

The  $K$  formulas are first developed here based on the results recently obtained by Sanders [1, 2]. As stated above, the  $K$  solutions for these loadings are not expressed as functions of a single geometric parameter. Sanders presented approximate formulas for the energy release rate for these loadings, which are readily converted into  $K$  formulas. The formulas are, in essence, functions of two geometric parameters for given elastic constants, which may be written in either of the following forms.

$$K = \sigma \sqrt{\pi(R\theta)} F(\lambda, \theta)$$

or

$$K = \sigma \sqrt{\pi(R\theta)} F\left(\theta, \frac{R}{t}\right)$$

(1)

where  $\sigma$  is an applied stress,  $2R\theta$  is the total circumferential length of through-crack.

In this discussion,  $\theta$  and  $R/t$  are chosen as geometric parameters and the second form of Eq. (1) is employed for the stress intensity expression. Approximate  $K$  formulas and the subsequent estimation formulas for the crack opening areas are developed specifically for  $R/t = 10$ , which is considered to be a typical value of interest in the present study. That is, the function  $F(\theta)$  in the subsequent discussion represents  $F(\theta; 10)$ .

Let  $P$  and  $M$  be the axial tensile force and bending moment, respectively, applied to the pipe far from the crack location and let subscripts  $t$  and  $b$  represent respectively tension and bending. The nominal stresses due to tension and bending are defined by

$$\begin{aligned}\sigma_t &= \frac{P}{2\pi R t} \\ \sigma_b &= \frac{M}{\pi R^2 t}\end{aligned}\tag{2}$$

The stress intensity factors are expressed in the following forms.

$$\begin{aligned}K_t &= \sigma_t \sqrt{\pi(R\theta)} F_t(\theta) \\ K_b &= \sigma_b \sqrt{\pi(R\theta)} F_b(\theta)\end{aligned}\tag{3}$$

where  $F_t(\theta)$  and  $F_b(\theta)$  are non-dimensional functions. The numerical values of the functions  $F_t(\theta)$  and  $F_b(\theta)$  are calculated from Sanders' approximate formulas for  $R/t = 10$ , which are tabulated as follows.

$(F_t(\theta)$  and  $F_b(\theta)$  for  $R/t = 10$ )

$\theta$	$F_t(\theta)$	$F_b(\theta)$
0°	1.000	1.000
9	1.039	1.037
18	1.151	1.140
27	1.314	1.278
36	1.505	1.425
45	1.725	1.580
54	1.987	1.747
63	2.305	1.934
72	2.702	2.154
81	3.209	2.406
90	3.872	2.760
99	4.764	3.209
108	6.003	3.827

These values represent slight overestimates of  $F_t(\theta)$  and  $F_b(\theta)$  [1,2]. The following approximate expressions of the functions  $F_t(\theta)$  and  $F_b(\theta)$  represent the values of the table with a reasonable accuracy (within a few percent).

$$F_t(\theta) = 1 + 7.5\left(\frac{\theta}{\pi}\right)^{3/2} - 15\left(\frac{\theta}{\pi}\right)^{5/2} + 33\left(\frac{\theta}{\pi}\right)^{7/2} \tag{4}$$

$$F_b(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{3/2} - 13.6\left(\frac{\theta}{\pi}\right)^{5/2} + 20\left(\frac{\theta}{\pi}\right)^{7/2}$$

$$(0 < \theta < 100^\circ)$$

When the pipe is subjected to axial force and bending moment at the same time, the total stress intensity factor is obtained simply by superposition of these separate factors.

$$K_{\text{total}} = K_t + K_b \quad (5)$$

The crack opening areas due to tension and bending,  $A_t$  and  $A_b$ , may be conveniently expressed in the following form.

$$A_t = \frac{\sigma_t}{E} (\pi R^2) I_t(\theta) \quad (6)$$

$$A_b = \frac{\sigma_b}{E} (\pi R^2) I_b(\theta)$$

where  $E$  is the Young's modulus, and  $I_t(\theta)$  and  $I_b(\theta)$  are non-dimensional functions.

The crack opening area for the tensile loading,  $A_t$ , is obtained by energy method (Castigliano's theorem) as follows:

$$A_t = \frac{1}{t} \frac{\partial U_t}{\partial \sigma_t} = 2 \int_0^\theta \frac{\partial}{\partial \sigma_t} \left( \frac{K_t^2}{E} \right) R d\theta \quad (7)$$

since

$$G = \frac{1}{Rt} \frac{\partial U_t}{\partial \theta} = \frac{K_t^2}{E} \quad (8)$$

where  $U_t$  is the total strain energy in the cracked pipe. Combining Eqs. (3), (6) and (7), the functions  $I_t(\theta)$  is obtained as follows:

$$I_t(\theta) = 4 \int_0^\theta \theta \{F_t(\theta)\}^2 d\theta \quad (9)$$

Substituting  $F_t(\theta)$  given by Eq. (4),  $I_t(\theta)$  is written as

$$I_t(\theta) = 2\theta^2 \left[ 1 + \left(\frac{\theta}{\pi}\right)^{3/2} \{8.6 - 13.3\left(\frac{\theta}{\pi}\right) + 24\left(\frac{\theta}{\pi}\right)^2\} \right. \\ \left. + \left(\frac{\theta}{\pi}\right)^3 \{22.5 - 75\left(\frac{\theta}{\pi}\right) + 205.7\left(\frac{\theta}{\pi}\right)^2 \right. \\ \left. - 247.5\left(\frac{\theta}{\pi}\right)^3 + 242\left(\frac{\theta}{\pi}\right)^4\} \right] \quad (10)$$

$$(0 < \theta < 100^\circ)$$

The crack opening area for bending load,  $A_b$ , however, can not be obtained as readily because the "crack absent stress distribution" is not uniform along the crack (direct application of the energy method is difficult). Therefore,  $A_b$  or  $I_b(\theta)$  will be estimated in the following way.

First, comparison of the crack absent stress distributions for tensile and bending loads, the following bounds are imposed on  $A_b$ :

$$A_t(\sigma_t = \sigma_b \cos\theta) < A_b(\sigma_b) < A_t(\sigma_t = \sigma_b) \quad (11)$$

$$(\cos\theta)I_t(\theta) < I_b(\theta) < I_t(\theta)$$

Where  $A_b(\sigma_b)$  is the crack opening area by bending, and  $A_t(\sigma_t = \sigma_b \cos\theta)$  and  $A_t(\sigma_t = \sigma_b)$  are the crack opening area due to axial force with tension stress  $\sigma_b \cos\theta$  and  $\sigma_b$ , respectively. The first approximation would be to take the

average uniform stress between these extremes and

$$A_b(\sigma_b) \simeq A_t(\sigma_b \frac{1 + \cos\theta}{2}) = A_t'(\sigma_b (\cos \frac{\theta}{2})^2)$$

or

(12)

$$I_b(\theta) = (\cos \frac{\theta}{2})^2 I_t(\theta)$$

Since the function  $I_b(\theta)$  given by Eq. (12) may yield underestimated values of the crack opening by bending, the stress intensity factors  $K_t$  and  $K_b$  are compared in a similar manner. Corresponding to Eq. (11), it is obvious that

$$K_t(\sigma_t = \sigma_b \cos\theta) < K_b(\sigma_b) < K_b(\sigma_t = \sigma_b)$$

or

(13)

$$(\cos\theta)F_t(\theta) < F_b(\theta) < F_t(\theta)$$

Averaging the extremes

$$F_b(\theta) = (\cos \frac{\theta}{2})^2 F_t(\theta)$$
(14)

Comparison of the numerical values of  $F_t(\theta)$  and  $F_b(\theta)$ , however, shows that Eq. (14) always underestimates  $F_b(\theta)$  and that the values of  $F_b(\theta)$  lie between the following two bounds

$$(\cos \frac{\theta}{2})^2 F_t(\theta) < F_b(\theta) < \frac{1 + (\cos \frac{\theta}{2})^2}{2} F_t(\theta)$$
(15)

Therefore, taking the following expression for  $I_b(\theta)$  instead of Eq. (14),

the risk of excessive underestimation of the crack opening area caused by bending load may be avoided

$$I_b(\theta) = \frac{1 + (\cos \frac{\theta}{2})^2}{2} I_t(\theta) = \frac{3 + \cos \theta}{4} I_t(\theta) \quad (16)$$

where  $I_t(\theta)$  is given by Eq. (10).

The total crack opening area caused by axial tension and bending can be written as,

$$\begin{aligned} A_{total} &= A_t + A_b \\ &\approx \frac{\sigma_t}{E} (\pi R^2) I_t(\theta) \left[ 1 + \frac{\sigma_b}{\sigma_t} \left( \frac{3 + \cos \theta}{4} \right) \right] \\ \text{or} \\ &\approx \frac{\sigma_b}{E} (\pi R^2) I_t(\theta) \left[ \frac{\sigma_t}{\sigma_b} + \frac{3 + \cos \theta}{4} \right] \end{aligned} \quad (17)$$

The effect of the yielding near the crack tip may be incorporated by the customary method of plastic zone corrections in which  $\theta$  in these formulas is replaced by  $\theta_{eff}$ .  $\theta_{eff}$  is obtained by using

$$\theta_{eff} = \theta + \frac{K_{total}^2}{2\pi R \sigma_Y^2} \quad (18)$$

for plane stress (maximum) plastic corrections. Repeated iterative procedures may be necessary for obtaining  $\theta_{eff}$ .



Circumferential Through-Crack Subjected to Internal Pressure

For a pipe subjected to internal pressure,  $p$ , the membrane stress,  $\sigma$ , in the axial direction is estimated by

$$\sigma = \frac{1}{2} \frac{pR}{t} \tag{19}$$

The stress intensity factor for a circumferential through-crack is normally expressed in the following form.

$$K_p = \sigma \sqrt{\pi a} \cdot F_p(\lambda) \tag{20}$$

where  $2a = 2R\theta$  is the total circumferential length of the crack,  $F_p(\lambda)$  is nondimensional function of  $\lambda = a/\sqrt{Rt}$  and the subscript  $p$  represents pressure loading. Contrary to the cases of axial force and bending load, the geometric factor  $F_p(\lambda)$  for this case is a function of a single geometric parameter as mentioned earlier.

The following formula empirically represents the curve of  $F_p(\lambda)$  presented in Rooke-Cartwright's work [3]. The approximate formula is, for convenience, expressed in a form consistent with the formula for longitudinal crack which will be subsequently discussed. Accuracy of the formula is within a few percent over the range specified.

$$\begin{aligned} F_p(\lambda) &= (1 + 0.3225\lambda^2)^{1/2} && (0 \leq \lambda \leq 1) \\ &= 0.9 + 0.25\lambda && (1 \leq \lambda \leq 5) \end{aligned} \tag{21}$$

where  $\lambda = a/\sqrt{Rt}$

Applying the energy method again, the crack opening area,  $A_p$ , is readily obtained as follows

$$A_p = \frac{\sigma}{E} (2\pi Rt) \cdot G_p(\lambda) \quad (22)$$

where  $G_p(\lambda)$  is given by an integral

$$G_p(\lambda) = 2 \int_0^\lambda \lambda \{F_p(\lambda)\}^2 d\lambda$$

Corresponding to Eq. (21),  $G_p(\lambda)$  is evaluated as

$$\begin{aligned} G_p(\lambda) &= \lambda^2 + 0.16\lambda^4 & (0 \leq \lambda \leq 1) \\ &= 0.02 + 0.81\lambda^2 + 0.30\lambda^3 + 0.03\lambda^4 & (1 \leq \lambda \leq 5) \end{aligned} \quad (23)$$

The effect of yielding near the crack tip may be similarly incorporated using the effective (plastic zone corrected) crack size which is calculated from the iterative relation

$$a_{\text{eff}} = a + \frac{K_p^2}{2\pi\sigma_y^2} \quad (24)$$

#### Longitudinal Through-Crack Subjected to Internal Pressure

For a pipe subjected to internal pressure,  $p$ , the hoop stress,  $\sigma$ , is estimated by

$$\sigma = \frac{pR}{t} \quad (25)$$

The stress intensity factor for a longitudinal through-crack of length  $2a$  is given by

$$K = \sigma\sqrt{\pi a} \cdot F(\lambda) \quad (26)$$

where again  $\lambda = a/\sqrt{Rt}$ .

The geometric factor  $F(\lambda)$  can be empirically expressed over the range of interest by

$$\begin{aligned} F(\lambda) &= (1 + 1.25\lambda^2)^{1/2} && (0 \leq \lambda \leq 1) \\ &= 0.6 + 0.9\lambda && (1 \leq \lambda \leq 5) \end{aligned} \quad (27)$$

Eq. (27) provides a good approximation for the shell factor  $F(\lambda)$  with accuracy of the order of one percent [3, 4, 5, 6].

The crack opening area,  $A$ , can be obtained by the method in the previous discussion.

$$A = \frac{\sigma}{E} (2\pi Rt) \cdot G(\lambda) \quad (28)$$

where  $G(\lambda)$  corresponding to Eq. (27) is given by

$$\begin{aligned} G(\lambda) &= \lambda^2 + 0.625\lambda^4 && (0 \leq \lambda \leq 1) \\ &= 0.14 + 0.36\lambda^2 + 0.72\lambda^3 + 0.405\lambda^4 && (1 \leq \lambda \leq 5) \end{aligned} \quad (29)$$

Iteration with a plastic zone correction similar to Eq. (24) can be applied to account for the yielding effect near the crack tip.

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555  
September 4, 1981

*E. McKenna*

Docket No. 50-244  
LS05-81-09-018

John E. Maier  
Vice President  
Electric and Steam Production  
Rochester Gas & Electric Corporation  
89 East Avenue  
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPIC III-5.8, PIPE BREAK OUTSIDE CONTAINMENT  
R. E. GINNA

The staff evaluation of SEP Topic III-5.8 was transmitted to you on June 24, 1980. This evaluation identified five staff positions for which an implementation schedule was requested. Your response was provided in a letter dated August 7, 1980.

Each of the five positions, your responses and staff resolutions are discussed below.

Staff Position 1

Because high and moderate energy line breaks in the Screen House could damage the power supplies to all service water pumps, the licensee must provide protection for these power supplies in accordance with Standard Review Plan 3.6.1 consistent with the service water system modifications which must be performed in connection with other ongoing SEP reviews and the fire protection review. Modifications to provide this protection can be acceptably delayed until the SEP integrated assessment of the plant provided that the diesel generator cooling method described, in the licensee's December 28, 1979 fire protection safe shutdown analysis, is tested to assure its timely availability and its capability to provide adequate cooling. The results of this testing should be submitted for NRC staff review.

Response to Staff Position 1

It is planned to conduct the alternative diesel generator cooling method test by June 1981.

### Resolution

The alternate diesel generator cooling method depends on installation of hose connections to each diesel generator. These connections have not yet been installed. As discussed in the June 1980 SER, protection should be provided for Buses 17 and 18 and associated cables. Such modifications should be coordinated in the integrated assessment with fire protection and other SEP topic concerns.

### Staff Position 2

The licensee must provide the means to warn the control room operator that flooding conditions exist in the Intermediate Building sub-basement. The licensee should provide the implementation schedule for this capability.

### Response to Staff Position 2

Based on RG&E's review of this scenario, we find the proposed solution to be unnecessary. Present routine walk-through inspections of the Intermediate Building would detect a pipe leak long before there were any danger of flooding safety-related equipment. If the postulated leak occurred at a level above the sub-basement, leakage into the sub-basement via the floor drains would be obvious during the routine once-per-shift walk-throughs. And even a large secondary side break would result in only a 2-foot depth in the sub-basement. If the leak were in the Service Water piping located in the sub-basement of the Intermediate Building, there would be a significant time interval between the initiation of the crack and the flooding of safety-related equipment. The Intermediate Building sub-basement has a volume of approximately 50,000 ft<sup>3</sup>. With a service water leak rate of about 585 gpm (as calculated on p. 13 of the NRC assessment), it would take over 10 1/2 hours to begin flooding the basement level. It does not seem conceivable that a sizeable leak rate such as this would not be detected, visibly or audibly by personnel during the walk-throughs, or by personnel monitoring the control board (the 585 gpm leak would be a significant fraction - 10% - of the Service Water pump flow).

### Resolution

The staff has determined from discussions with the licensee during a site visit on June 2, 1981, that there are two sump pumps in the sub-basement. Operation of the pumps is alarmed at the water treatment station. A control room alarm is provided indicating that an alarm condition exists at the water treatment station. As stated in the topic evaluation, even



if the basement elevation was flooded, safe shutdown would not be prevented. Based on this, and the other information provided above, the staff concludes that there are adequate means to warn of flooding conditions in the sub-basement and therefore, that no modifications are required.

### Staff Position 3

Based on our evaluation of Main Steam (MS) and Main Feed (MF) line breaks in the Turbine Building and Intermediate Building, the licensee should (1) proceed with the design and installation of jet impingement shielding in the Intermediate Building (as previously committed to by the licensee), (2) provide protection from the effects of the failure of the Turbine Building/Intermediate Building cinder block wall for the MS atmospheric dump valves and assess the need for and provide protection as necessary for the MS safety valves. The installation of additional jet impingement shielding for the MS bypass valves and associated piping is not necessary since the bypass valves are not required for safe shutdown or pipe break mitigation. A proposal to accommodate item (2) above should be submitted for staff review.

### Response to Staff Position 3

Protection from the effects of the Turbine Building/Intermediate Building cinder block wall failure on the atmospheric dump valves and main steam safety valves will be integrated into the modification program resulting from RG&E's review of I&E Bulletin 80-11, "Masonry Wall Design." Our initial response to this bulletin is contained in a July 7, 1980 letter from L. D. White, Jr. (RG&E) to Mr. Boyce H. Grier (NRC Region I Director).

### Resolution

Additional information in response to I&E Bulletin 80-11 was submitted by the licensee on November 4, 1980 and January 30, 1981. The SEP review of these letters has revealed that pipe break loads were not included in this evaluation of masonry wall design. Furthermore, since the evaluation against original design criteria showed that the walls would satisfy their intended function, no assessment of effects of cinder block wall failure has been provided. Therefore, the licensee should comply with item 2 above.

### Staff Position 4

Since certain moderate energy line breaks (MELB) in the mechanical equipment room could result in flooding both battery rooms, the licensee must provide protection from the effects of these postulated MELB's in accordance with the acceptance criteria of Standard Review Plan 3.6.1. The licensee should provide a schedule for the implementation of this position.



Response to Staff Position 4

It is presently planned to separate the battery rooms from the mechanical equipment room, where the source of a Service Water leakage exists, by replacing the doorway with a watertight wall. This modification should be completed by June 1981.

Resolution

The modification will be completed shortly. The licensee also plans to install at the same time a means of removing water from the mechanical equipment room into the turbine building. The staff concludes that these modifications will adequately mitigate the effects of these postulated MELB's.

Staff Position 5

- .. To preclude adverse environmental conditions resulting from a heating steam or CVCS letdown break in the Auxiliary Building, the licensee must analyze the adequacy of once-per-shift inspections to prevent the formation of the adverse environment or to provide some other acceptable means of preventing the existence of the adverse environment. The results of this analysis (with a commitment to provide the required protection, if necessary) should be submitted for NRC staff review.

Response to Staff Position 5

RG&E is performing an evaluation to determine the effects of a CVCS letdown or steam heating line break in the Auxiliary Building in the vicinity of safety-related equipment. The results of this study and proposed modifications, will be submitted to the NRC for review in January 1981. Pending the resolution of any noted concerns, present once-per-shift inspections, together with the procedures available for isolation of the steam heating line, should provide adequate protection against the effects of significant adverse environment damaging safety-related equipment.

Resolution

The environmental effects of these breaks on safety-related equipment are being addressed as part of Unresolved Safety Issues (USI) "Qualification of Class 1E Equipment". Per the Commission's Memorandum and Order of May 23, 1980, all safety-related electrical equipment must be qualified for the adverse environments they would experience by June 30, 1982. Therefore, this item will not be further addressed under Topic III-5.8.

The staff now considers this SEP topic to be completed except for completion of the commitments discussed above and of modifications necessary to protect equipment in the screen house and Turbine Building/Intermediate Building.

Enclosed is the revised evaluation which will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

*Dennis M. Crutchfield*  
Dennis M. Crutchfield, Chief  
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Enclosure:  
As stated

cc w/enclosure:  
See next page

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SEP REVIEW  
OF  
PIPE BREAK OUTSIDE CONTAINMENT  
TOPIC III-5.B  
FOR THE  
R. E. GINNA NUCLEAR POWER PLANT

## INTRODUCTION

The safety objective of Systematic Evaluation Program (SEP) Topic III-5.8, "Pipe Break Outside Containment" is to assure that pipe breaks would not cause the loss of needed functions of safety-related systems, structures and components and to assure that the plant can be safely shut down in the event of such breaks. The needed functions of safety-related systems are those functions required to mitigate the effects of the pipe break and safely shutdown the reactor plant. The current criteria for review of pipe breaks outside containment are contained in Standard Review Plan 3.6.1 and 3.6.2 including their attached Branch Technical Positions.

## BACKGROUND

In December 1972, the staff sent letters (Reference 1) to all power reactor licensees requesting an analysis of the effects of postulated failures of high energy lines outside of containment. A summary of the criteria and requirements in this letter is set forth below:

- a. Protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from all effects resulting from ruptures in pipes carrying high energy fluid, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig, respectively, up to and including a double-ended rupture of such pipes. Breaks should be assumed to occur in those locations specified in the "pipe whip criteria." The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement), and environmental.
- b. In addition, protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying fluid routed in the vicinity of this equipment. The size of the cracks should be assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

In response to our letter and subsequent requests for additional information, Rochester Gas and Electric (RG&E, the licensee) submitted a report, "Effects of Postulated Pipe Breaks Outside the Containment Building," and several additional letters providing information and schedules for plant modifications. A complete bibliography of these letters is contained in the NRC Safety Evaluation Report (SER) for Amendment No. 29 for the Ginna plant (Ref. 2). The SER for Amendment No. 29 also provides the NRC staff evaluation of certain facility modifications proposed by the licensee to provide protection from the effects of a postulated pipe break outside containment. Reference 3 approved the licensee's augmented Inservice Inspection (ISI) Program which is intended to ensure a very low probability of pipe breaks at locations in the main steam and main feed systems where modifications to mitigate the effects of the

breaks could not be installed. In addition, the licensee committed to make certain modifications in conjunction with the Systematic Evaluation Program (SEP) reevaluation of the effects of pipe breaks outside containment.

The NRC staff reevaluation of the effects of pipe breaks outside containment under SEP Topic III-5.B involves the comparison of the Ginna plant with current criteria for pipe breaks outside containment. The staff used an "effects oriented" approach to determine the acceptability of plant response to pipe breaks, i.e., each structure, system, component, and power supply which must function to mitigate the effects of the pipe break and to safely shutdown the plant was examined to determine its susceptibility to the effects of the postulated break. Break effects considered were compartment pressurization, pipe whip, jet impingement, spray, flooding, and environmental conditions of temperature, pressure, and humidity. This review complements that of SEP Topic III-12, "Environmental Qualification of Safety-Related Equipment."

(The effects of potential missiles generated by fluid system ruptures and rotating machinery will be evaluated under SEP Topic III-4C, "Internally Generated Missiles.")

The previous evaluation of pipe breaks outside containment for the Ginna Plant was performed using some methods and criteria which are no longer used by the staff in the review of current plants. For example, the current definition of a high energy fluid system as one that is maintained under conditions where either or both the maximum operating temperature and pressure exceeds 200°F and 275 psig is different from the definition applied in the previous review where a high energy fluid system was one in which both temperature and pressure exceed 200°F and 275 psig. The SEP reevaluation of this topic was performed using the criteria extracted from Standard Review Plan 3.6.1 and 3.6.2 and their attached Branch Technical Positions.

Data for this assessment was gathered during a visit to the Ginna plant on September 25-27, 1979.

The staff issued its draft evaluation of this topic on June 24, 1980 (Reference 14). The licensee was requested to provide a schedule for resolution of the open items. The licensee response was transmitted by letter dated August 7, 1980 (Reference 15). This information has been incorporated into this evaluation.

#### EVALUATION

The results of the SEP reevaluation of pipe breaks outside containment for the Ginna plant are provided in Table 1. The table lists the zones within the plant which contain systems required for safe shutdown and/or systems required to mitigate the effects of postulated pipe breaks. These zones are the screen house, diesel generator rooms, intermediate building (elev. 293', 278' and 253'), turbine building (elev. 289', 271', and 253'), control room, relay room, battery rooms, mechanical equipment room, and auxiliary building (elev. 271', 253', and 235').

The safe shutdown systems which were examined from the standpoint of protection from pipe break effects are identified in the SEP Safe Shutdown Review for Ginna (Ref. 9). These systems are:

- (a) Reactor Protection System
- (b) Auxiliary Feed System
- (c) Main steam safety, isolation, and atmospheric dump valves
- (d) Service Water System
- (e) Chemical and Volume Control System
- (f) Component Cooling Water System
- (g) Residual Heat Removal System
- (h) Instrumentation for Shutdown and Cooldown
- (i) Emergency Power (AC and DC) and control power for the above systems and components.

This section provides additional information used to evaluate certain pipe breaks listed in Table 1.

#### Screen House

Service Water System (SWS) or fire system Moderate Energy Line Breaks (MELB's) and heating steam line breaks could result in the loss of the SWS by damaging 480V electrical buses 17 and 18 or their associated electrical cabling. Loss of the SWS would result in a plant trip because of the loss of several components cooled by the SWS such as the reactor feed pump lube oil systems, circulating water pumps, and the CCW system. In accordance with current criteria, a pipe break which results in a reactor or turbine trip results, in turn, in a loss of offsite power. To supply AC power following a loss of offsite power, redundant emergency diesel generators are available; however, the diesel generators are supplied cooling water by the SWS. Therefore, the postulated pipe break could cause the total loss of AC power at the plant, and reactor core decay heat removal would be dependent on the turbine driven auxiliary feed pump which is susceptible to a postulated single active failure.

The licensee has been evaluating the SWS in connection with the ongoing NRC fire protection review and the SEP reviews of flooding and tornado missiles. To conduct a plant cooldown following a fire which causes a loss of all SWS with no offsite power available, the licensee has developed a procedure which is described in Ref. 4. The procedure requires the installation of fire hoses from the city hydrant system to provide the diesel generators cooling water and to provide additional water to the auxiliary feed pumps for steam generator makeup water. While the fire hoses are being installed, the turbine driven auxiliary feed pump is used to add water from the Condensate Storage Tank to the steam generators for decay heat removal. After a diesel generator is operable, additional auxiliary feed pumps and the reactor coolant system charging pumps can be operated as required. According to the procedure, fire hoses and portable pumps would have to be connected to one CCW heat exchanger if a plant cooldown to cold shutdown conditions were required with no SWS flow available.

The proposed procedure could be used for the pipe break case even if the turbine driven auxiliary feed pump is assumed to fail. Without feedwater addition, the steam generators can remove decay heat for approximately 50 minutes before they are boiled dry. This time could be used to makeup the temporary diesel generator cooling connections to start a diesel generator and a motor driven auxiliary feed pump.

The staff's conclusion and position for resolution of these postulated pipe breaks in the Screen House and their associated equipment failures are contained in the CONCLUSIONS section of this report.

#### Intermediate Building Flooding

As noted in several places in Table 1, flooding from pipe breaks in the Intermediate Building (IB) would flow via open stairways and hatch gratings to the sub-basement of the IB. Sufficient drainage area is available so that no appreciable buildup of water would occur on any floor of the IB except for the sub-basement. No equipment necessary for safe shutdown or flood mitigation is located on this level; but, if the flooding condition went unchecked, the IB 253' elevation could be affected to a depth of about 30 inches. Equipment on this elevation includes the auxiliary feed pumps and the reactor trip breakers. If this equipment were flooded, a reactor trip would occur and the auxiliary feed system would be inoperable. The standby auxiliary feed system, which is not located in the IB, would still be operable even if a loss of offsite power occurred. Operation of the sump pumps in the sub-basement is alarmed at the water treatment station. In the control room an alarm is provided to alert the operator of an alarm condition at the water treatment station.

#### Intermediate Building Main Steam and Main Feed Breaks

Postulated Main Steam (MS) and Main Feed (MF) system High Energy Line Breaks (HELB's) in the IB could result in the following:

- (a) The "A" MS line on the 293' elevation could damage cable trays 16, 72, and 122 by jet impingement. At this elevation, these trays contain control and power cables for the containment fan coolers and the containment purge exhaust fans. These systems are not required to function to mitigate a MS break outside containment or to shutdown the plant.
- (b) The 30" dia. "A" MS line on the 293' elevation could damage the north IB cinder block wall (whip or impingement), an interior steel column supporting the IB floors above 293' (whip), or the cable trays discussed in (a) above (whip or impingement).
- (c) On the 278' elevation of the IB, large MS line breaks could damage both the floor supporting the MS header and MF line "A"; and a break at the juncture of the 36" dia. and the 30" dia. steam lines could overstress the anchors which connect the lines to the IB structure.
- (d) A "B" MF line break on the 278' elevation could damage one or more steam safety valves for the "A" steam generator.





- (e) IB pressurization by a large HELB was predicted in Ref. 6, for the bounding case of the 36" dia. MS line break, to result in failure of the cinder block walls and roof beams and decking of the IB although the IB structure was not predicted to be damaged.
- (f) In Reference 6, it is stated that a "8" 30" dia. MS line break outside the IB at the penetration to the containment building could damage the control building by means of pipe whip.

Because of the severe consequences of these postulated MS and MF line breaks in the IB and because plant modifications to prevent these consequences were not practical (Ref. 7), the licensee undertook a two-part program to reduce the vulnerability of the plant to a HELB in the IB. The first part of the program was an augmented radiographic inspection program, described in Ref. 8, to provide added assurance that postulated large MS and MF breaks would not occur. This program was reviewed and accepted by the NRC staff in 1975 (Ref. 3). The second part of the licensee's program was to move essential equipment from the IB into locations unaffected by an HELB in the IB. The intent of this program is to preclude the large (greater than the equivalent of six inch diameter) breaks and acceptably mitigate the small breaks. A summary of plant modifications installed and equipment relocated is provided in Ref. 2.

The licensee has committed to install additional modifications in conjunction with the SEP review of this issue. These modifications would include the installation in the IB of jet impingement shielding for one steam generator atmospheric dump valve and all MS safety valves. In the Intermediate Building, the licensee committed to install jet impingement protection for the two main steam bypass valves and associated piping. The staff has concluded that the installation of jet impingement shielding for the MS bypass valves and associated piping is not necessary since the bypass valves are not required for safe shutdown or pipe break mitigation. Also, modifications to the IB cinder block wall resulting from the analysis of HELB's in the Turbine Building will be made as necessary upon completion of the SEP. The licensee's commitment is detailed in Ref. 10.

A comparison of the IB pressurization caused by a 6" dia. HELB provided in Ref. 6 with the design limits of the IB cinder block wall provided in Ref. 11 shows that even this small HELB could fail the cinder block wall. As a result of this failure, equipment in the Turbine Building could be damaged. The only equipment which may be of concern from the standpoint of plant shutdown are the MF regulating valves and bypass valves on the 270' elevation of the Turbine Building. However, even if these valves were damaged, the Standby Auxiliary Feed System (SAFS) would be available to feed the steam generators and effect a safe shutdown of the plant. The SAFS was installed to provide steam generator feed in case a pipe break in the IB damaged the Auxiliary Feed System.

#### Turbine Building Main Steam and Main Feed Breaks

Postulated MS and MF system HELB's in the Turbine Building (TB) could result in:

- (a) The 24" MS lines could whip into the IB wall at the proper elevation to damage the "B" MS line safety valves, atmospheric dump valve, and steam supply line to the turbine driven auxiliary feed pump.
- (b) MS and MF breaks could pressurize the TB itself. The following pressures have been calculated:

Location	Breaks			
	20" MF @ 270'	24" MS @ 298'	36" MS @ 270'	12" MF @ 270'
TB 298'	.456 psi	.589 psi	.742 psi	.233 psi
TB 270' and 243'	.848 psi	.507 psi	1.26 psi	.259 psi

These results are provided in Ref. 11 for the 20" MF break and in Ref. 12 for the other breaks. The pressurization of the TB could adversely affect those areas adjacent to the TB in which safe shutdown or pipe break mitigating equipment is located. These areas are the control room, diesel generator room, relay room, battery room and the IB.

Again, because of the consequences of these postulated MS and MF line breaks in the TB, the licensee utilized the two-part program to reduce the vulnerability of the plant to these HELB's. The licensee's previously approved augmented inspection program has been applied in the TB to MS lines larger than 12" dia. and several locations on the 20" dia. MF header. The inspection program limits the breaks which must be considered to a 12" MS or a 20" MF line break which are the largest potential double-ended breaks in locations which are not inspected. Of these, the 20" MF is more limiting. To protect the areas adjacent to the TB from the effects of HELB's, the licensee has installed pressure diaphragm walls between the TB and the control room, relay room, battery rooms, mechanical equipment room, and diesel generator rooms. The design differential pressure for these walls is 0.7 psi for the control room and 1.14 psi for the other spaces. The NRC evaluation of these walls is in Reference 2.

The pressure resulting from a 20" MF or 12" MS line break in the TB is sufficient to cause failure of the TB/IB cinder block walls (design pressure .13 psid). If these walls failed, the following systems and components could be damaged by falling cinder blocks or adverse environmental conditions: one containment purge exhaust fan on the IB 298' elev., the auxiliary feed system (AFS) steam supply valves on the IB 278' elev., and the AFS turbine driven pump, reactor trip breakers, and reactor rod control motor generator sets on the IB 253' elev.

The purge exhaust fan is not required to function to mitigate a HELB outside containment. The rod control motor generators and reactor trip breakers fail safe if damaged and would not prevent a reactor trip (core shutdown). The AFS function is required for a safe shutdown; however, the SAFS has been installed

by the licensee to accomplish this function if a HELB disables the AFS. The turbine driven AFS pump is not specifically required to operate following a postulated HELB since, even if offsite power were assumed to be lost, the redundant emergency diesel generators would be available to power the two SAFS pumps or the remaining two AFS pumps all of which are driven by electric motors. Only one of these four motor driven pumps is required for a plant shutdown and cooldown.

The discussion in the previous paragraph shows that most of the equipment which can be damaged by a failure of the TB/IB block walls is not necessary for HELB mitigation or safe plant shutdown. However, the MS isolation valves and MS safety and atmospheric relief valves are necessary for HELB mitigation and safe shutdown. Although the safety valves would probably not be rendered inoperable by failure of the TB/IB walls, the licensee will be requested to assess this possibility and consider incorporating protection of the valves with the jet impingement shields to be installed. Both atmospheric dump valves would have to be protected from the effects of the wall failure.

#### Battery Room/Mechanical Equipment Room Flooding

A SWS or fire system MELB in the mechanical equipment room could flood both battery rooms and result in a loss of all emergency DC power. A 20" diameter SWS line break in the mechanical equipment room would result in a calculated flooding rate of 585 gpm using the methods of Ref. 5. No sump level or flood alarms are installed in this space or in the battery rooms which are connected to the mechanical equipment room by normally closed non-watertight doors. The licensee has committed to replace the non-watertight doors by a wall.

#### Auxiliary Feed System Breaks on the 253' Elevation of the IB

The AFS discharge lines from the pumps in the IB (253' elev.) to the "B" MF header run along the north wall of the IB at approximately the 270' elevation. A break in this line, which is a high energy line, could result in pipe whip or jet impingement on cable trays and containment electrical penetrations in that area. (The steam lines for the turbine driven AFS pump are also in this area but are not considered high energy lines since they are not pressurized during normal plant conditions.) Reference 4 presents an analysis of plant shutdown capability following an exposure fire in this area which destroys all electrical cables and equipment in the area. This condition envelopes the damage which could be done by the AFS HELB. To provide safe shutdown capability following the fire, the licensee has proposed methods and identified plant modifications to be installed (Ref. 4). Upon completion of these modifications and because of previously installed modifications, specifically the standby AFS and relocation of safe shutdown instruments from the IB, the plant will have an acceptable level of protection from the effects of AFS breaks on the 253' elevation of the IB.

## CONCLUSIONS

Based on the information submitted by the licensee and obtained during the site visit to the Ginna plant, we have determined that the following review areas have not been adequately addressed in previous staff safety evaluations and should be resolved with the SEP:

1. SWS and fire system MELB's and heating steam line breaks in the screen house could result in the loss of all SWS flow, by damaging Buses 17 and 18, and the loss of all AC power. The licensee is implementing a method to provide cooling to the onsite emergency diesel generators which is not dependent on the SWS. The staff position regarding these pipe breaks is that the licensee must provide protection for Buses 17 and 18 and their associated cables from the effects of the breaks in accordance with Standard Review Plan 3.6.1 consistent with the modifications which must be performed on the SWS to accommodate other ongoing SEP reviews, e.g., the tornado missile and fire protection reviews.
2. Based on our evaluation of Main Steam (MS) and Main Feed (MF) line breaks in the Turbine Building and Intermediate Building, the licensee should provide protection from the effects of the failure of the Turbine Building/Intermediate Building cinder block wall for the MS atmospheric dump valves and assess the need for and provide protection as necessary for the MS safety valves. The proposal should be submitted for staff review.

TABLE 1. EFFECTS OF PIPE BREAK OUTSIDE CONTAINMENT

<u>Zone</u>	<u>Pipe Break</u>	<u>Affected Mitigating System</u>	<u>Affected Safe Shutdown System</u>	<u>Adequacy of Protection/Remarks</u>
Screen House	SWS (MELB)* or Fire System (MELB)	None	SWS Power Supply Bus 17 & 18	Potentially inadequate. Spray from a SWS or fire system leak can affect both and cause loss of all SWS pumps. See discussion in EVALUATION section.
	CW (MELB)	None	SWS, Bus 17 & 18	Adequate. Previously analyzed in CW flooding evaluation (Ref. 13).
	Heating steam (HELB)	None	SWS Power Supply Bus 17 & 18	Potentially inadequate. High temperature environment effects on cables to Buses 17 and 18 could cause loss of SWS. See remarks above.
Diesel Generator Room 1A (253')	SWS (MELB) or Fire System (MELB)	None	Diesel generator 1A	Adequate. Spray from MELB may affect generator or associated electrical panels, but redundant diesel generator and offsite power are available as backup. Flooding in room is detected by sump pump alarm in control room. Cable vault below diesel generator room is protected from flooding by watertight manhole cover.
	Heating steam (HELB)	None	Diesel generator 1A	Adequate. Fire protection temperature detector warn control room of high temperature conditions. No LOP; other diesel available.
Diesel Generator Room 1B (253')	SWS (MELB) or Fire System (MELB)	None	Diesel generator 1B	Adequate. See comments above for MELB in 1A diesel room. SWS supply line to diesel 1A passes through 1B diesel room but leakage from a crack break in this SWS line would not be enough to render the 1A diesel inoperable through loss of cooling water.

\*A list of abbreviations is provided at the end of this table.

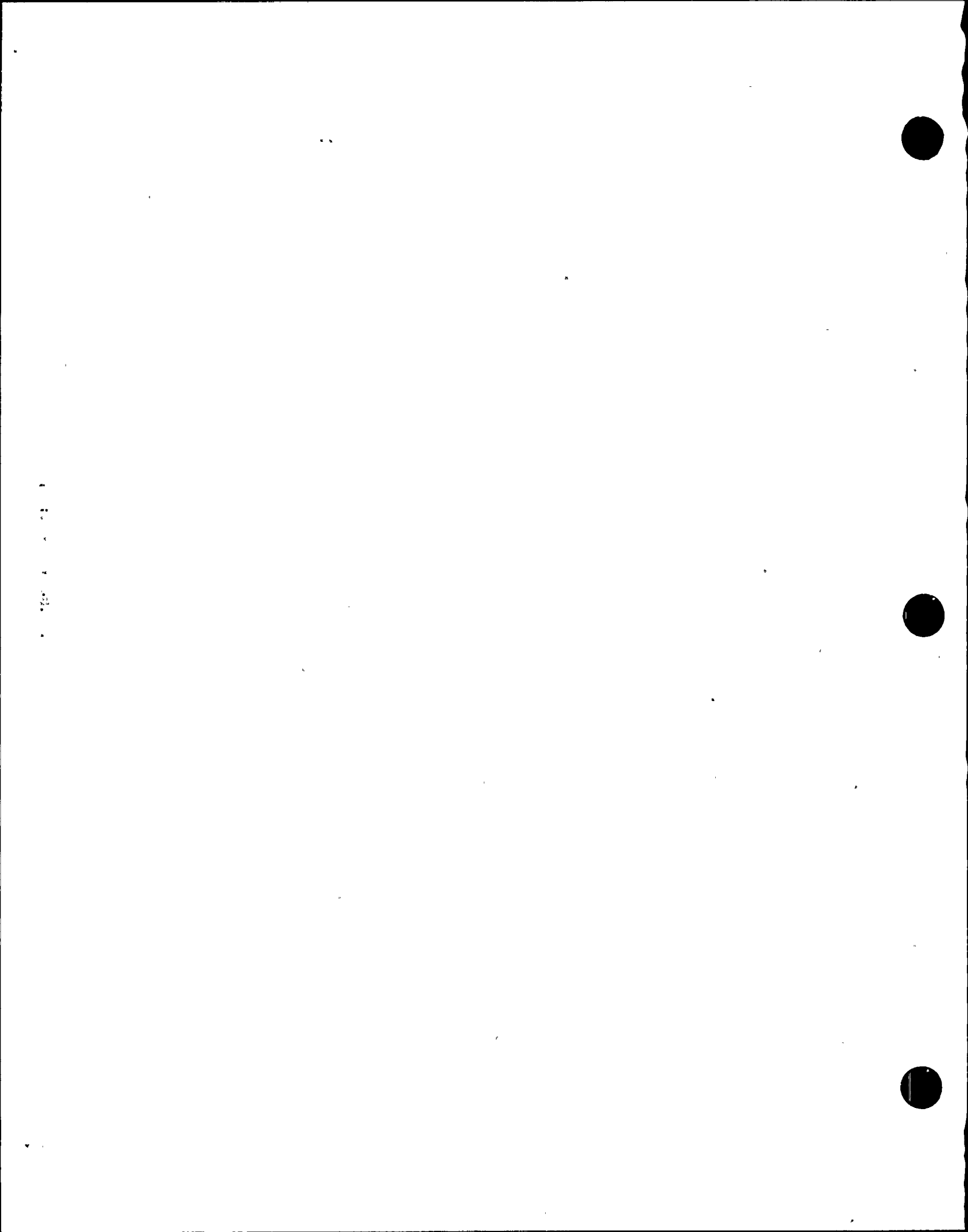


TABLE 1. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection/Remarks
Diesel Generator Room 1B (253') (continued)	Heating Steam (HELB)	None	Diesel generator 1B	Adequate. See remarks above for heating steam leak in diesel room 1A.
Intermediate Building (293')	Fire system (MELB)	None	None	Adequate; see evaluation section of this report.
	MS and MF (HELB) [crack breaks]	Various; see evaluation section of this report.	Various; see evaluation section of this report.	Adequate. Jet impingement from a crack in "A" MS line could impact cable trays 16, 72, 122. Although these trays are safety related, at the 293' elevation they contain no cables needed to mitigate the effects of the break or to safely shutdown. Environmental effects of MS and MF crack breaks would be experienced throughout the intermediate building. Licensee has modified the plant to withstand these conditions in the intermediate building and to prevent these conditions from spreading to the auxiliary building. (Refs. 2 and 10)
	MS, "A" and "B" headers, MF, "B" header (HELB) [large break]	Various, see evaluation section of this report.	Various; see evaluation section of this report.	Adequate. Although a large MS or MF line rupture in the intermediate building has the potential to structurally damage the building, these ruptures are effectively precluded by the licensee's ongoing inservice inspection of these lines.



TABLE 1. (Continued)

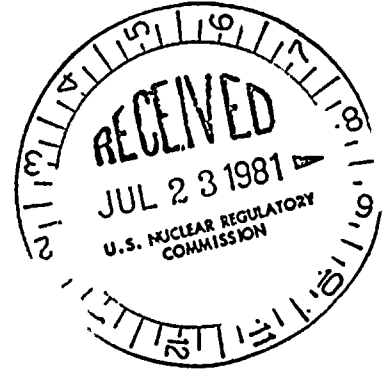
Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection/Remarks
Intermediate Building (278') (continued)	AFS (HELB)	None	MSIVs, Atmospheric MS dump valves, MS safety valves, AFS turbine driven pump steam supply valve	Adequate. Jet impingement of AFS water (~80°F) on MSIVs, safety valves would not render these inoperable. Impingement on AFS turbine driven pump steam supply valve or either atmospheric dump air control system could render these inoperable; however, the turbine driven pump is not normally used for safe shutdown and the function of steam generator makeup can be performed by other AFS and SAFS pumps, and the atmospheric dumps can be manually operated by handwheels.
Intermediate Building (278')	Fire system (MELB)	None	None	Adequate. See remarks above for MELB at 293' elevation.
	MS and MF (HELB) [crack break]	Various; see evaluation section of this report.	Various; see evaluation section of this report.	Adequate. Licensee has protected or moved instrumentation required to mitigate the effects of the breaks or to safely shutdown. This is further discussed in the evaluation section of this report.
	MS and MF (HELB) [large break]	Various; see evaluation section.	Various; see evaluation section.	Adequate. See remarks for large MS or MF line break on 293' elevation of intermediate building.

\*Some protection will be installed in conjunction with the SEP review of the facility.



July 21, 1981

Docket No. 50-244  
LS05-81-07-070



Mr. John E. Maier, Vice President  
Electric and Steam Production  
Rochester Gas & Electric Corporation  
89 East Avenue  
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SEP TOPICS II-1.B, POPULATION DISTRIBUTION AND  
III-4.D, SITE PROXIMITY MISSILES - R. E. GINNA

Enclosed are the staff's final evaluations of SEP Topics II-1.B and III-4.D for the R. E. Ginna Nuclear Power Plant. These evaluations are based on our review of your topic safety assessment reports submitted by letters dated April 15, 1981 and April 16, 1981, respectively.

You will note that we have revised your calculated population density which is more properly obtained by dividing the total population within a given distance by the total area of the complete circle (including both level and water) whose radius is the distance of interest.

This completes our evaluation of Topics II-1.B and III-4.D.

These evaluations will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

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DSu use (07)*

Dennis M. Crutchfield, Chief  
Operating Reactors Branch No. 5  
Division of Licensing

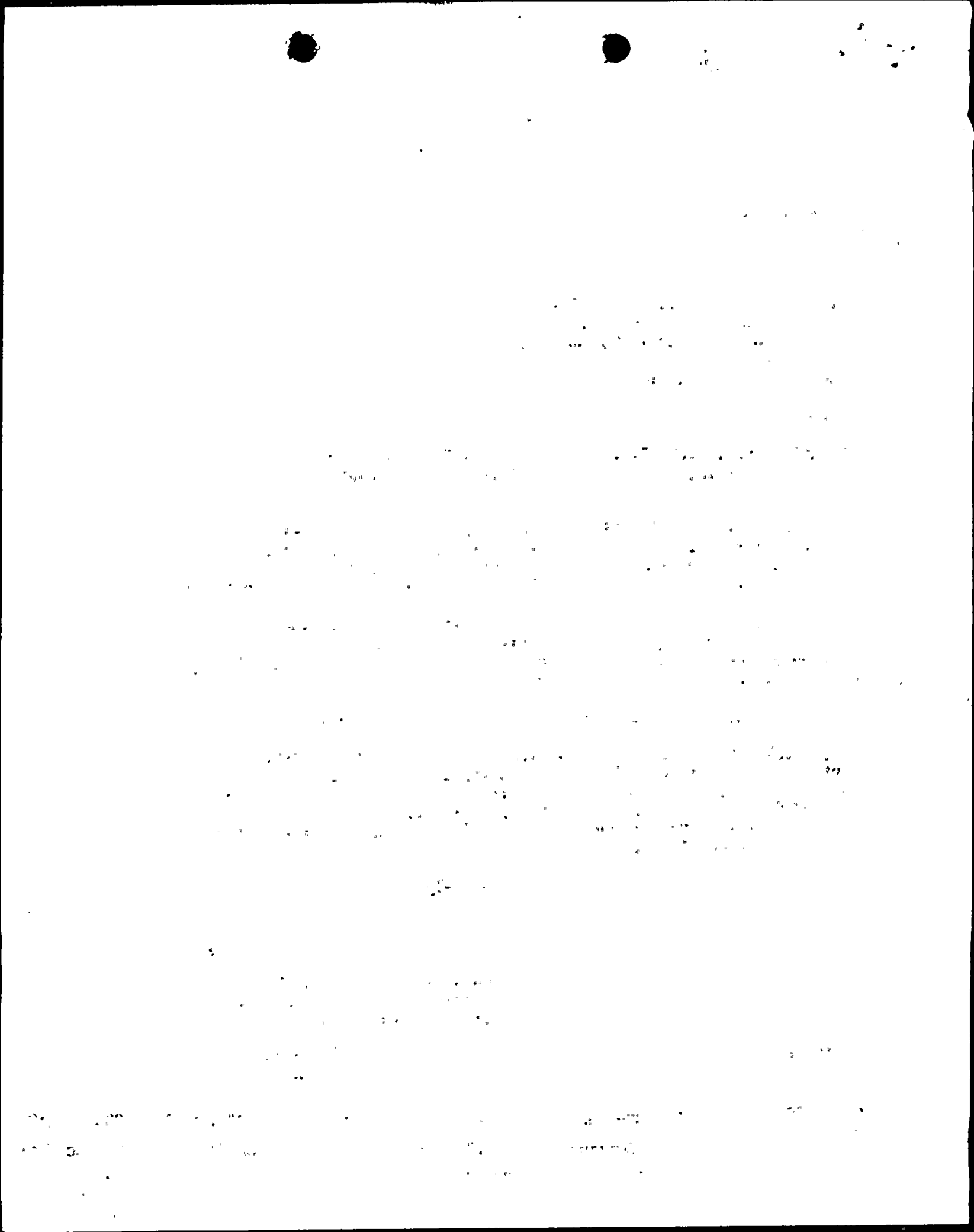
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Enclosure:  
As stated

OFFICE	cc w/enclosure: See next page	SEPBB:DL	SEPBB:DL <i>CWB</i>	SEPBB:DL <i>Wm</i>	ORB#5:DL:PM	ORB#5:DL:C
SURNAME		GCvna:dk	CBerlinger	WRussell	RSnaider	DCrutchfield
DATE		7/16/81	7/16/81	7/16/81	7/20/81	7/20/81



Mr. John E. Maier

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R. E. GINNA  
SYSTEMATIC EVALUATION PROGRAM TOPIC  
II-1.B, POPULATION DISTRIBUTION

I. INTRODUCTION

The safety objective of this topic is to ensure that the previously-established low population zone and population center distance specified for the site are compatible with the current population distribution, and are in accordance with the guidelines of 10 CFR Part 100.

II. REVIEW CRITERIA

Sections 100.10 and 100.11 of 10 CFR Part 100, "Reactor Site Criteria" provides the site evaluation factors which should be considered when evaluating sites for nuclear power reactors. These sections include guidelines for determining the exclusion area, low population-zone and population center distance.

III. RELATED SAFETY TOPICS

Topic II-1.A, reviews the licensee's control over the exclusion area. Various other topics will evaluate the capability of the plant to meet the dose criteria of 10 CFR Part 100 at the exclusion area boundary and low population zone. The adequacy of emergency preparedness planning for the area surrounding the plant including the low population zone is being assessed by the Commission in a separate review effort.

IV. REVIEW GUIDELINES

The review has been conducted in accordance with Standard Review Plan (SRP) Section 2.1.3, "Population Distribution."

V. EVALUATION

The R. E. Ginna site is in the township of Ontario, in the northwest corner of Wayne County, New York, on the north shore of Lake Ontario about 20 miles ENE of the center of the City of Rochester and 40 miles WSW of Oswego. The land surrounding the site is primarily of an agrarian nature and sparsely populated. There are no substantial population centers, industrial complexes, transportation arterials, parks, or other recreational facilities within a three mile radius of the Ginna site.<sup>2</sup> The City of Rochester is the largest population center within a 50 mile radius of the site (241,539 people, with 701,745 in the metropolitan area<sup>7</sup>). The nearest community with a population of 1,000 or more is the Town of Ontario with its center located about 3½ miles from the site. The preliminary estimated 1980 census for the Town of Ontario is 7,452.<sup>7</sup>

To develop the Wayne County and Monroe County Radiological Emergency Response Plans for the R. E. Ginna Nuclear Power Station, a recent survey of the population within a five-mile radius was completed. Figure J-2 from the Wayne County Radiological Response Plan, reproduced as Figure 1

of this evaluation, details the population within 5 miles of Ginna, based on preliminary 1980 population estimate. RG&E estimates that 10,864 persons reside within five miles of the plant, a density of 138 persons per square mile averaged over the entire area. (It should be noted that this figure compares favorably with the 1980 population projection of 10,934 persons shown in Figure 2.4-2 of the Ginna FSAR, which was published in 1968).

Other than the residents of the area, there are no large groups of transients within five miles of the site. The only parks near the site are Webster Beach Park in Monroe County, approximately 6 miles west of the plant site, and B. Forman Park in Wayne County, approximately 8 miles east of the plant site. There are no federal recreational facilities in the area. There are no state parks, public campsites, or special use areas within ten miles of the plant.<sup>2</sup> Wayne County does have a migrant labor population, primarily for apple picking, during the June-October season. Approximately 115 farmworker camps of five or more persons are scattered throughout Wayne County<sup>8</sup>, with a total population of about 4400 migrants. Information from Rural New York Farmworker Opportunities shows that there are only 12 camps, with about 130 migrants, located in the vicinity of the Ginna site.<sup>10</sup>

The nearest population center to the Ginna site containing more than 25,000 residents is the "Rochester urbanized area," whose eastern boundary is about ten miles from the site.<sup>2</sup> The only other population center of more than 25,000 persons is the City of Auburn (population 32,442),<sup>7</sup> located more than 40 miles SE of the site.



The low population zone specified for the Ginna site is the area within a 3 mile (4,827 meter) radius of the plant.<sup>9</sup> A review of current population estimates and projected growth estimates indicate that the population growth in the area since the plant received an operating license in 1969 has been modest, and this trend is expected to continue. No population center of 25,000 residents has developed, or appears likely to develop, closer than the eastern boundary of the Rochester urbanized area.

VI. CONCLUSION

The staff concludes that the low population zone and population center distances specified for the Ginna site is in conformance with the requirements of 10 CFR Part 100 in that the population center distance is more than one and one-third times the distance from the reactor to the outer boundary of the low population zone (10 miles vs. 3 miles).

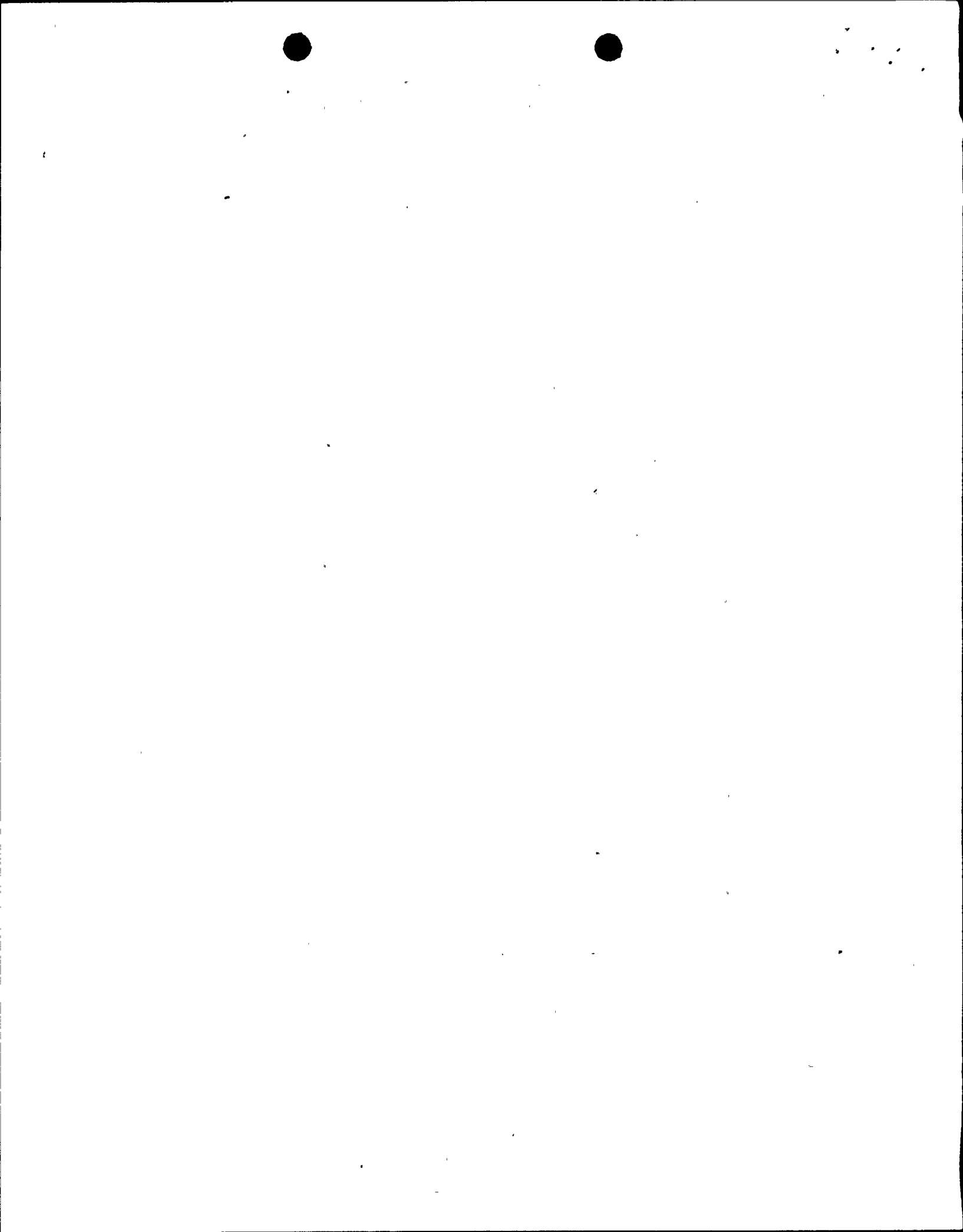
We further conclude that the site conforms to the current licensing criteria. This completes the evaluation of SEP Topic II-1.B for the Ginna site.

VII. REFERENCES

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.4.
2. Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Section 2.1.3, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Wayne County Radiological Emergency Response Plan, Draft Rev. B, November 1980.
6. Monroe County Radiological Emergency Response Plan Draft, Rev. B, November 1980.
7. Preliminary Report, 1980 Census of Population and Housing, New York, published by the Bureau of the Census, U. S. Department of Commerce, February 1981.
8. Conversation with the New York State Health Department, April 13, 1981.
9. Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission in the Matter of Rochester Gas and Electric Corporation Robert Emmett Ginna Nuclear Power Plant Unit No. 1, Docket No. 50-244 (SER), Section 2.1, June 19, 1969.
10. Letter, Thomas J. Harris, RNYFO, to George Wrobel, RG&E, April 10, 1981.

11. Rochester Gas and Electric Corporation, Ginna Nuclear Station  
Radiation Emergency Plan, Proposed January 1981.
12. New York State Radiological Emergency Preparedness Plan,  
December 1980.





R. E. GINNA  
SYSTEMATIC EVALUATION PROGRAM TOPIC  
TOPIC III-4.D, SITE PROXIMITY MISSILES  
(INCLUDING AIRCRAFT)

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for a site proximity missile.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis." of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic II-1.C, "Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial and Military Facilities" provides a description of the potential missile hazards.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.3, "Evaluation of Potential Accidents," 3.5.1.5, "Site Proximity Missiles (except Aircraft)," and 3.5.1.6, "Aircraft Hazards."

V. EVALUATION

The potential for hazardous activities in the vicinity of the Ginna plant has been addressed in SEP topic II-1.C, "Potential Hazards due to Industrial, Transportation, Institutional and Military Facilities". As indicated therein, there is little industrial activity near the plant. The distances to the nearest land transportation routes are such (about 1700 feet to the nearest highway, and 3 1/2 miles to the nearest railroad) that the risk associated with potential missiles from transportation accidents on these routes are within the SRP 2.2.3 guidelines. Similarly, the nearest large gas pipelines are about six miles from the plant, and do not pose a missile threat to the plant. Major Lake Ontario shipping routes are also sufficiently far away (about 23 miles) so as not to present a credible missile hazard from lake traffic. There are no military facilities or activities near the plant which would create a missile hazard.

The review of SEP Topic II-1.C also evaluated the potential for aircraft becoming a missile hazard, both in connection with the operation of the Williamson Flying Club Airport, which is about ten miles ESE of the plant, and due to commercial air traffic in and out of Rochester via federal airways V2N and V2, which are 2 1/2 and 10 miles from the plant site.

As evaluated in Topic II-1.C, it was determined that, since the Williamson Flying Club Airport expected a maximum of only 5000 operations per year, and is about 10 miles from the site, the criteria in III.3.a and III.3.b of SRP 3.5.1.6 were met, and there is no need to determine the probability of an aircraft crash into the plant. Further, the hazard to the plant from commercial aircraft use of airways V2 and V2N was shown to be only  $5.1 \times 10^{-8}$  and  $1.4 \times 10^{-8}$  per year, respectively. No danger to the plant from commercial airline traffic is thus expected.

#### Conclusion

Since current regulatory criteria are met with respect to SEP Topic III-4.D, "Site Proximity Missiles", it can be concluded that this topic is complete for the R. E. Ginna site. No additional review for this topic is required during the SEP integrated assessment.



VI. REFERENCES

1. Rochester Gas and Electric Corporation, Robert Emmett Ginna Nuclear Power Plant Unit No. 1 - Final Facility Description and Safety Analysis Report (FSAR), Sections 2.2 and 2.5.
2. Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant Unit No. 1, Environmental Report, Volume 1, Sections 2.1 and 2.2.
3. Nuclear Regulatory Commission NUREG-75/087, Standard Review Plan, Sections 2.2.1, 2.2.2, 2.2.3, and 3.5.1.6, September 1975.
4. Code of Federal Regulations, Section 10, Part 100 (10 CFR 100).
5. Sterling Power Project Nuclear Unit No. 1, Preliminary Safety Analysis Report Addendum, Rochester Gas and Electric, Volume 1, Sections 2.1 and 2.2.
6. U.S. Nuclear Regulatory Commission Regulatory Guide 1.91, Rev. 1, February 1978.
7. Letter, John E. Maier, RG&E, to Dennis M. Crutchfield, NRC, SEP Topic II-1.C, "Potential Hazards Due to Transportation, Industrial, Institutional and Military Facilities", April 15, 1981.