ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

NEW YORK STATE

JOHN E. MAIER
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

October 16, 1981

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topic XV-9, Startup of an Inactive Loop
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

We have reviewed the draft topic evaluation for SEP Topic XV-9, which was provided by your letter dated August 26, 1981, and concur that the assessment accurately represents the as-built condition of Ginna.

Very truly yours,

John E. Maier

J. E. Maier

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
August 26, 1981

Docket No. 50-244
LS05-81-08-059

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U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Mr. Maier:

SUBJECT: SEP TOPIC XV-9, STARTUP OF AN INACTIVE LOOP
R. E. GINNA

Enclosed is the draft topic evaluation for XV-9. This evaluation compares your facility with the criteria currently used by the regulatory staff for licensing new facilities.

Please inform us if you as-built facility differs from the licensing basis assumed in our assessment within 30 days of receipt of this letter.

This evaluation will be a basic input to the integrated safety assessment. for your facility unless you identify changes.needed to reflect 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
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. John E. Maier

cc

Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Bureau of Nuclear Operations
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town of Ontario
107 Ridge Road West
Ontario, New York 14519

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Mr. Thomas B. Cochran
Natural Resources Defense Council, Inc.
1725 I Street, N. W.
Suite 600
Washington, D. C. 20006

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

R. E. GINNA

TOPIC XV-9 - STARTUP OF AN INACTIVE LOOP

I. INTRODUCTION

The startup of an inactive coolant loop is examined to assure that the introduction of cooler or deborated water into the core does not lead to an unacceptable loss of fuel clad integrity or overpressurization of the primary system. The guidelines used are contained in SRP Section 15.4.4, Startup of an Inactive Loop at an Incorrect Temperature.

II. EVALUATION

The startup of an inactive loop at power results in a core reactivity increase when the colder water of the idle loop reaches the core. Reactor protection for this event is provided mainly by administrative procedure and the inherent stability of the core.

Technical Specification 3.1.1.1 permits operation up to 130 Mwt (8.5% of rated power) with only one reactor coolant pump operating. An orderly power reduction is required to below 8.5% power if a pump is lost while operating at a higher power level. Above 50% power loss of a reactor coolant pump will cause a direct reactor trip.

The licensee provided an analysis of this event in the FSAR (Reference 1). An analog computer representation of the primary system was used.

The initial conditions and assumptions used in the analysis include:

1. inactive loop is 20°F cooler than the active loop;

2. large negative moderator temperature coefficient;

3. small doppler coefficient;

4. high heat transfer coefficient between primary and secondary so that the cold leg of the inactive loop is at saturation temperature for the steam generator secondary;

5. instantaneous start of idle pump; and

6. mixing with the active loop flow in the plenum.

Although no uncertainty was applied to the initial power level, these assumptions are judged to be sufficiently conservative for this analysis. The 20°F ΔT was selected on the basis that at the low power level permitted for one-loop operation, the difference between the temperature in the inactive loop and the temperature in the active loop will not be great. The effect of the other assumptions, such as the temperature coefficient, is to augment the reactivity and power increase caused by the cold slug.

Results of the analysis plotted in the Ginna FSAR indicate a temperature and pressure decrease of 10°F and 30 psi, respectively. The power increased, but a pressure spike was not generated because of coolant contraction due to the temperature decrease. The reactor stabilizes without actuation of any automatic protection features or dependence on operator action. The relatively minor perturbations in system parameters, and the reduced power condition of the core ensure that the operating limits are not exceeded.

The effects of startup of an inactive loop are less severe than those due to a small steam line break with one loop operable, which has been analyzed by Westinghouse as shown in Reference 2. To accommodate the steam line break, Technical Specification 3.1.1.1.b requires that a higher shutdown margin be maintained for one-loop operation.

Operation with less than all loops in service is the subject of SEP Topic IV-1.A, which was completed by Reference 3.

III. CONCLUSION

As a part of the SEP review of Ginna, the analysis of Startup of an Inactive Loop at an Incorrect Temperature has been evaluated against the criteria of SRP Section 15.4.4. Based on this evaluation, we have concluded that the consequences of startup of an inactive loop have been adequately addressed by the licensee and the acceptance criteria are satisfied.

REFERENCES

1. "R. E. Ginna Nuclear Power Plant Final Safety Analysis Report," Section 14.1.7; September 1969.

2. Leboeuf, Lamb, Leiby and MacRae, for Rochester Gas and Electric Corporation, "Application for Amendment to Operating License;" September 22, 1975.

3. Letter to D. Ziemann (NRC) to L. White (RG&E); May 29, 1979.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 1, 1980

Docket No. 50-244

Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. White:

By letter dated November 16, 1979, we sent you the list of topics which would not be reviewed by your plant in the Systematic Evaluation Program (SEP). These topics were deleted from the SEP review because they were not applicable to your plant or they were being reviewed as part of an ongoing generic issue outside the SEP. The following topics should be added to the November 16 listing. Removal of these additional topics from the SEP review results from 1) related generic NRC activities and 2) non-applicability of topics to certain facilities. The following is a list of such topics and indicates their disposition:

Topic	Categorization	Disposition
II-2.B	Generic	Compliance with Appendix E is being evaluated under the Emergency Prepardness Program Office (EPPO) effort. Compliance with Appendix I is being evaluated as part of the NRC Appendix I review effort.
II-2.D	Generic	Compliance is being evaluated under the EPPO effort.
III-8.B	Deleted	This topic was originally intended for BWR's.
VI-8	Generic	Will be reviewed by NRC as part of TMI Task Action Plan (NUREG-0660) as specified in May 7, 1980 letter to all operating reactor licensees.
XV-11	Deleted	This item applies specifically to BWR's.
XV-18	Deleted	This item applies specifically to BWR's. The radiological consequences of steam line breaks for PWR's is part of Topic XV-2.

The following two topics will not be reviewed by SEP in their entirety due to overlap with other NRC activities:

Topic	Disposition
II-1.B	Those portions dealing with plant emergency plans will be evaluated as part of the EPPO effort.
X	Those portions of the auxiliary systems being reviewed as part of Lessons Learned, the TMI Action Plan and IE Circular 80-04 will not be reviewed by SEP.

In addition, the original DBE review list indicated that Group VI events, "Uncontrolled Rod Withdrawal at Power" and "Uncontrolled Rod Withdrawal - Low Power Startup" were part of Topic XV-13 which does not apply to PWR's. The actual topic reference should be Topic XV-8, "Control Rod Misoperation" which is applicable to PWR's. The new topic reference list for Topic XV-8 should include Standard Review Plans 15.4.1 and 15.4.2.

Enclosure 1 to this letter includes a revised listing of the DBE Accident and Transient Groups applicable to PWR's. Of note should be the addition of Group XII, "Steam Generator Tube Failure and the inclusion of Fuel Handling Accidents in Group VIII.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

cc w/enclosure:
Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Rochester Committee for Scientific Information
Robert E. Lee, Ph.D.
P. O. Box 5236 River Campus Station
Rochester, New York 14627

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Technical Development Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town of Ontario
107 Ridge Road West
Ontario, New York 14519

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Thomas B. Cochran
Natural Resources Defense Council, Inc.
1725 I Street, N. W.
Suite 600
Washington, D. C. 20006

Richard E. Schaffstall
KMC, Incorporated
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

PWR DBE TOPIC LIST

Accident and Transient Groups

	RELATED TOPIC	DESIGN BASIS EVENT	SRP
Group I (PWR)	XV-1	Decrease in feedwater temperature	15.1.1
	XV-1	Increase in feedwater flow	15.1.2
	XV-1	Increase in steam flow	15.1.3
	XV-1	Inadvertent opening of steam generator relief/safety valve	15.1.4
	XV-9	Start-up of inactive loop	15.4.4
	XV-10	System malfunction causing decrease in boron concentration	15.4.6
Group II	XV-3	Loss of external load	15.2.1
	XV-3	Turbine trip	15.2.2
	XV-3	Loss of condenser vacuum	15.2.3
	XV-3	Steam pressure regulatory failure	15.2.5
	XV-5	Loss of feedwater flow	15.2.7
	XV-6	Feedwater system pipe break	15.2.8
Group III	XV-2	Steam line break inside containment	15.1.5
	XV-2	Steam line break outside containment	15.1.5
Group IV	XV-4	Loss of AC power to station auxiliaries	15.2.6
	XV-24	Loss of all AC power	-
Group V	XV-7	Loss of forced coolant flow	15.3.1
	XV-7	Primary pump rotor seizure	15.3.3
	XV-7	Primary pump shaft break	15.3.4
Group VI	XV-8	Uncontrolled rod assembly with-drawal at power	15.4.2
	XV-8	Uncontrolled rod assembly with-drawal low power start-up	15.4.1
	XV-8	Control rod misoperation	15.4.3
	XV-12	Spectrum of rod ejection accidents	15.4.8
Group VII	XV-19	Spectrum of loss of coolant accidents	15.6.5
Group VIII	XV-21	Drop of cask or heavy equipment	15.7.5
	XV-20	Radiological Consequences of Fuel Damaging Accidents (inside and outside containment)	15.7.4
Group IX	XV-15	Inadvertent opening of PWR pressurizer relief valve	15.6.1
Group X	XV-14	Inadvertent operation of ECCS or CVCS malfunction that causes an increase in coolant inventory	15.5.1 15.5.2

Enclosure 6

	RELATED TOPIC	DESIGN BASIS EVENT	SRP
Group XI		Not applicable to PWRs	
Group XII	XV-17	Steam Generator Tube Failure	15.6.3

TOPIC XV-12(SYSTEMS)

SEE TOPIC XV-1

FOR SYSTEMS EVALUATION

TOPIC XV-12(DOSES)

SEE TOPIC XV-2

FOR DOSE EVALUATION

TOPIC XV-13

SEE TOPIC II-4.E

TOPIC XV-16

SEE TOPIC XV-2

FOR DOSE EVALUATION

TOPIC XV-17 (SYSTEMS)

SEE TOPIC XV-1

FOR SYSTEMS EVALUATION

TOPIC XV-17 (DOSES)

SEE TOPIC XV-2
FOR DOSE EVALUATION

TOPIC XV-18

SEE TOPIC II-2.B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 27, 1981

Docket No. 50-244
LS05-81-10-001

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: R. E. GINNA - SEP TOPIC XV-19 (SYSTEMS) LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

On July 27, 1981 we transmitted to you a draft assessment of SEP Topic XV-19 (systems). In your letter of September 15, 1981 you provided comments in the form of a revised topic assessment.

The staff has evaluated the suggested revisions and we consider that they provide substantial additional information, but do not alter the staff's conclusions. Therefore, we will use your revised assessment (enclosed) as our final evaluation.

We now consider Topic XV-19 (systems) to be complete. The enclosed safety evaluation will be a basic input to the integrated safety assessment for your facility. The 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,

Richard [illegible] for
Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

Mr. John E. Maier

cc

Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N. W.
Suite 1100
Washington, D. C. 20036

Mr. Michael Slade
12 Trailwood Circle
Rochester, New York 14618

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Bureau of Nuclear
Operations
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town
of Ontario
107 Ridge Road West
Ontario, New York 14519

Resident Inspector
R. E. Ginna Plant
c/o U. S. NRC
1503 Lake Road
Ontario, New York 14519

Mr. Thomas B. Cochran
Natural Resources Defense Council, Inc.
1725 I Street, N. W.
Suite 600
Washington, D. C. 20006

U. S. Environmental Protection Agency
Region II Office
ATTN: Regional Radiation Representative
26 Federal Plaza
New York, New York 10007

Herbert Grossman, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

R. E. GINNA

SEPTEMBER, 1981

TOPIC XV-19: LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

I. INTRODUCTION

The capability of the R. E. Ginna Emergency Core Cooling System to mitigate the consequences of a spectrum of Loss of Coolant Accidents (LOCAs) is evaluated to assure that pipe breaks in the reactor coolant system (RCS) do not result in a loss of core cooling capability. Detailed acceptance criteria for Emergency Core Cooling System (ECCS) performance are contained in 10 CFR 50.46 and in Standard Review Plan Sections 15.6.5, 6.3 and supporting appendices. The five main criteria for acceptance are:

1. Peak clad temperature less than 2200°F
2. Maximum cladding oxidation less than 17%
3. Total hydrogen generation less than 1% of total zirconium in the active fuel region
4. Maintenance of coolable geometry
5. Long term coolability

A spectrum of break sizes up to and including a double-ended break of the largest pipe at various break locations is examined using an approved evaluation model which conforms to the requirements of Appendix K to 10 CFR 50 to verify that the acceptance criteria are met for a variety of postulated loss of coolant accidents.

II. EVALUATION

The Ginna power plant ECCS provides emergency core cooling water at three delivery pressures. The high pressure safety injection (HPI) system delivers borated water at up to 1400 psi (see Fig. 2-1 of Ref. 8 for SI pumped flowrate as a function of RCS pressure asuming 5% degradation from design head). This is different than current PWR's which use safety grade charging pumps which are capable of injection at operating pressure, about 2235 psi. Intermediate pressure passive injection is provided by the accumulators which are held at 700 psi by nitrogen

gas overpressure. The HPI system and the accumulators discharge into lines to each cold leg. Low pressure cooling water from the refueling water storage tank is delivered by the residual heat removal (RHR) system which becomes available at 140 psi. The low pressure injection flow is pumped directly into the upper plenum of the reactor vessel through two separate nozzles. More complete descriptions of these systems are provided in the Ginna safe shutdown report, (Ref. 1), and in the Final Safety Analysis Report (Ref. 7). Switchover from injection to recirculation mode is covered in SEP Topic VI-7.B.

The Ginna core currently contains 117 fuel assemblies designed and fabricated by Exxon Nuclear Company and 4 mixed oxide fuel assemblies designed and fabricated by Westinghouse Electric Corporation.

Analysis for the large pipe breaks was performed by Exxon Nuclear for the Cycle 8 fuel reload in Reference 2, with the staff evaluation presented in Reference 3. The limiting large break was reanalyzed in January 1980 (Ref. 9) to include an NRC model for fuel clad swelling and the incidence of fuel clad rupture.

The LOCA evaluation for the 4 Westinghouse assemblies is presented in Refs. 10 and 11 with NRC approval in Ref. 12.

The effect of the low pressure injection point being the vessel upper plenum instead of the cold legs is addressed in SEP Topic VI-7.A.2. This topic has been deleted from consideration in SEP since it is generic.

The small break analysis was performed by Westinghouse for Ginna during the initial Appendix K reviews (Ref. 8). Since the small breaks were clearly demonstrated to be non-limiting, later reloads re-evaluated only the large break spectrum.

In response to the NRC's Bulletins and Orders Task Force, additional small break analyses were performed on a generic basis. The Westinghouse calculations of Reference 4 were reviewed by the staff in Reference 5.

Large Break Analysis

The cycle 8 fuel reload safety analysis (Reference 2) examined six different pipe breaks in the cold leg. Hot leg breaks were not examined. Three double area guillotine breaks with discharge coefficients (C_D) of 1.0, 0.6, and 0.4 were analyzed. The other three breaks

considered were split breaks with discharge coefficients of 1.0, 0.6, and 0.4. The selection of breaks for this analysis was justified, based on previous evaluations, which clearly identified the cold leg split and guillotine breaks as the most limiting.

The assumptions and computer codes used in the LOCA analyses are covered in Reference 2. Some of the more important assumptions include:

1. Initial power at 102%
2. Reactor trip is neglected for large breaks
3. All accumulator water bypasses core until termination of bypass
4. Linear Heat Generation Rate of 13.76 kw/ft
5. Total peaking factor is 2.32
6. Fuel at beginning of life (BOL) conditions

These and the other assumptions used for these analyses were in accordance with 10 CFR 50.46 and Appendix K, 10 CFR Part 50 and have been shown (Ref. 2) to result in conservatively high peak cladding temperatures.

Results

The limiting case of the six breaks examined in the ECCS analysis for Ginna presented in Ref. 2 was the double-ended cold leg guillotine break with C_D=0.4. The peak clad temperature predicted was 1922°F, considerably below the 10 CFR 50.46 limit of 2200°F. Clad oxidation (peak and total) was also well within limits. It should be noted that guillotine breaks with a discharge coefficient smaller than 0.4 are not required in accordance with Reference 6. The analyses to determine the effect of using the NRC's model for fuel clad swelling and the incidence of clad rupture was performed using the models described in References 13-16. As described in Ref. 9, the revised model for fuel clad swelling and the incidence of rupture resulted in a peak clad temperature increase of 1°F for Exxon Nuclear fuel. Thus, analyses presented in Ref. 2 remain valid.

Small Break Analysis

As discussed above, plant-specific small break analyses were not performed by Exxon Nuclear because it had been shown in previous Westinghouse analyses for Ginna (Ref. 8) that the small breaks would not be the limiting case. Westinghouse analysis yielded a limiting small break

size of 4 inch diameter with a peak clad temperature of 1688°F.

Small Break Analyses - Post TMI

Generic analyses of small break LOCAs were submitted by the Westinghouse Owners Group in response to NRC Bulletins and Orders Task Force requirements. The staff has accepted these analyses as a basis for providing information on plant response and as an aid to developing guidelines for operator action. The generic analyses included consideration of the reduced head HPI system. The staff considers these generic analyses to be representative of the response for Ginna to a postulated small break LOCA.

Results - Small Break - Post TMI

As a result of the review of these analyses, the staff expressed concern about the applicability of current evaluation models and their application to the expanded scope of small break LOCA analyses now being considered. As part of the TMI Task Action Plan, which is beyond the scope of the SEP review, Westinghouse is to revise and resubmit the small break analysis methods for staff approval. Plant specific calculations, using these revised methods will then be required to show compliance with 10 CFR 50.46. These analyses should place special emphasis on accidents which actuate the HPI.

III. CONCLUSION

The loss of coolant accidents analyzed for the Ginna nuclear power plant meet the acceptance criteria.

New small break LOCA analyses using revised evaluation models will be conducted as part of the TMI Task Action Plan and will not be included as part of the SEP review.

The impact of upper plenum low head safety injection is being conducted by review of SEP Topic VI-7.A.2, "Upper Plenum Injection" and NRR Generic Task D-05. It is thus not included as part of this SEP topic.

REFERENCES

1. Safe Shutdown Systems for R. E. Ginna Nuclear Power Plant, SEP Topic VII-3, May 13, 1981.

2. "ECCS Analysis for R. E. Ginna Reactor with ENC WREM-II PWR Evaluation Model." Exxon Nuclear Company Report, XN-NF-77-58, December 1977.

3. "R. E. Ginna Nuclear Power Station Cycle 8 Reload Safety Evaluation Report." May 1, 1978.

4. "Report on Small Break Accidents for Westinghouse NSSS System" Westinghouse Nuclear Energy Systems Report, WCAP-9600, June 1979.

5. "Generic Evaluation of Feedwater Transients and Small Break Loss of Coolant Accidents in Westinghouse Designed Operating Plant", NUREG-0611, January 1980.

6. Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K, October 15, 1974 and the Supplement, dated November 13, 1974.

7. "R. E. Ginna Nuclear Power Plant Final Safety Analysis Report" September 1969.

8. Application dated September 3, 1974 and submitted September 6, 1974 from RG&E to the NRC.

9. Letter dated January 10, 1980 from L. D. White, Jr., RG&E to Dennis L. Ziemann, USNRC re ECCS Models.

10. Application dated December 14, 1979 and submitted December 20, 1979 from RG&E to the NRC.

11. Letter dated February 20, 1980 from L. D. White, Jr., RG&E to Dennis L. Ziemann, NRC.

12. Amendment No. 32 to the Ginna license transmitted by letter dated April 15, 1980 from Dennis L. Ziemann, NRC, to L. D. White, Jr., RG&E.

13. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA," XN-NF-78-30, August 1978.

14. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," XN-75-41:

    a. Volume I, July 1975
    b. Volume II, August 1975
    c. Volume III, Revision 2, August 1975

d. Supplement 1, August 1975
e. Supplement 2, August 1975
f. Supplement 3, August 1975
g. Supplement 4, August 1975
h. Supplement 5, Revision 5, October 1975
i. Supplement 6, October 1975
j. Supplement 7, November 1975

15. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II," XN-76-27, July 1976; Supplement 1, September 1976; Supplement 2, November 1976.

16. "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Updated ENC WREM-IIA: Responses to NRC Request for Additional Information," XN-NF-78-30(A) & XN-NF-78-30, Amendment 1(A), May 1979.