REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8109220618 DOC.DATE: 81/09/15 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
MAIER,J.E. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards revised draft assessment of SEP Topic XV-19 re LOCA resulting from spectrum of postulated piping breaks within RCPB,per NRC 810727 ltr.

DISTRIBUTION CODE: A035S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 7
TITLE: SEP Topics

NOTES:1 copy:SEP Sect. Ldr. 05000244

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
ACTION:	ORB #5 BC	04	7	7				
INTERNAL:	A/D MATL&QUAL	13	1	1	CONT SYS A	07	1	1
	HYD/GEO BR	10	2	2	~~I&E~~	06	2	2
	OR ASSESS BR	11	1	1	REG FILE	01	1	1
	SEP BR	12	3	3				
EXTERNAL:	ACRS	14	16	16	LPDR	03	1	1
	NRC PDR	02	1	1	NSIC	05	1	1
	NTIS		1	1				

SEP 25 1981

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~38~~ 39 ENCL ~~38~~ 39

RG

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

RG AND E

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

LAKE ONTARIO NEW YORK STATE

JOHN E. MAIER
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

September 15, 1981

RECEIVED SEP 22 1981 U.S. NUCLEAR REGULATORY COMMISSION

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-19 (Systems), Loss of Coolant Accident
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

We have reviewed the NRC draft assessment for topic XV-19 which was provided by your letter dated July 27, 1981 and have several comments. To facilitate your review, we have attached a revised draft assessment which incorporates our comments.

Very truly yours,

John E Maier
J. E. Maier

A035
S
1/1

R. E. GINNA

DRAFT ASSESSMENT, REV. 1
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.

<u>Large Break Analysis</u>

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

[illegible] into [illegible]

[illegible] directly into [illegible]

[illegible] difficult [illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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^{O}F, considerably below the 10 CFR 50.46 limit of 2200^{O}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^{O}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.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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.