Docket No. 50-244

MAY 14 1975

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14604

Gentlemen:

The Commission has issued the enclosed Amendment No. 7 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant (the facility). This amendment includes Change No. 16 to the Technical Specifications and is in response to your requests dated October 31, 1974; March 11, 1975; April 28, 1975; and May 13, 1975.

The amendment (1) changes operating limits in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR §50.46; (2) terminates restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License, and imposes instead, limitations established in accordance with 10 CFR §50.46; (3) incorporates an updated inservice inspection program for safety related components to (a) meet Section XI of the ASME Boiler and Pressure Vessel Code, (b) provide andaugmented inservice inspection program for high energy piping outside of containment, and (c) provide requirements for steam generator inspection consistent with Regulatory Guide 1.83; and (4) decrease the maximum permissible steam generator leakage from 1 gpm to 0.1 gpm to avoid operation with significant steam generator tube cracks.

You will note that the Technical Specifications include a requirement to remove D.C. power from motor operated valves 896A and 896B with the valves in the open position. We have agreed with this concept (in lieu of removal of A.C. power from these valves) with the understanding that the necessary modifications will be completed as soon as practicable. In this regard, we request that you furnish a description of your planned modifications for our review prior to making the change.

In your letter of May 13, 1975, you proposed interim methods (A.C. power removal and specific operator actions) to assure proper positioning of certain motor-operated valves despite high water level following a postulated LOCA. Within 30 days of the receipt of this letter, please provide us with your proposal for a permanent modification that will assure proper positioning of post-LOCA flooded valves.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the facility in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level, and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, §51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by:
Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 7
2. Negative Declaration
3. Environmental Impact Appraisal
4. Safety Evaluation
5. Federal Register Notice

cc w/enclosures:
See next page

DISTRIBUTION
Docket File
NRC PDR
Local PDRs
ORB#1 Reading
OELD
OI&E(3)
ACRS(14)
KRGoller
RSBoyd
AGiambusso
SVarga
SKari
TJCarter
RAPurple
TBAbernathy
JRBuchanan
RBevan
ZRRostecki
RMaccary
VStello
JSaltzman
NDube
CHebron
PCollins
BScharf(15)
BJones(4)
AESteen
JMMcGough
SMSheppard
TVWambach
VLRooney(2)

TR
CWB
5/14/75

SEE PREVIOUS YELLOW FOR OTHER CONCURRENCES(TR)

OFFICE	DRL:ORB#1	OELD	DRL:ORB#1	DRL:OR	DRL	DRL/D
SURNAME	VLRooney:dc	SCINTO	RAPurple	KRGoller	SVarga	AGiambusso
DATE	5/14/75	5/14/75	5/14/75	5/14/75	5/14/75	5/14/75

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14604

Gentlemen:

The Commission has issued the enclosed Amendment No. 7 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant (the facility). This amendment includes Change No. 16 to the Technical Specifications and is in response to your requests dated October 31, 1974, March 11 and April 28, 1975.

The amendment (1) changes operating limits in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR Section 50.46; (2) terminates restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License, and imposes instead, limitations established in accordance with 10 CFR Section 50.46; (3) incorporates an updated inservice inspection program for safety related components to (a) meet Section XI of the ASME Boiler and Pressure Vessel Code, (b) provide an augmented inservice inspection program for high energy piping outside of containment, and (c) provide requirements for steam generator inspection consistent with Regulatory Guide 1.83; and (4) decrease the maximum permissible steam generator leakage from 1 gpm to 0.1 gpm to avoid operation with significant steam generator tube cracks.

You will note that the Technical Specifications include a requirement to remove D.C. power from motor operated valves 896A and 896B with the valves in the open position. We have agreed with this concept (in lieu of removal of A.C. power from these valves) with the understanding that the necessary modifications will be completed as soon as practicable. In this regard, we request that you furnish a description of your planned modifications for our review prior to making the change.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the facility in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 7
2. Negative Declaration
3. Environmental Impact Appraisal
4. Safety Evaluation
5. Federal Register Notice

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	TJCarter
ORB#1 Reading	PCollins
NRC PDR	SVarga
Local PDR	CHebron
Attorney, OELD	ACRS (14)
OI&E (3)	AESteen
NDube	JRBuchanan
BJones (4)	TBAbernathy
JMMcGough	EP PM
JSaltzman	TR
SMSheppard	
TVWambach	
RAPurple	
SKari	
BScharf (15)	

L. A. for R. Maccary

OFFICE	RL:ORB#1	TR CHB	OELD	RL:OR	RL	RL
SURNAME	VLRooney:1b RAPurple			KRGoller	SVarga	AGiambusso
DATE	5/08/75	5/08/75	5/ /75	5/8/75	5/ /75	5/ /75

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc w/enclosures:
Arvin E. Upton, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, NW
Washington, D. C. 20036

Mr. Michael Slade
1250 Crown Point Drive
Webster, New York 14580

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

J. Bruce MacDonald, Deputy
Commissioner and Counsel
New York State Department of
Commerce
99 Washington Avenue
Albany, New York 12210

Lyons Public Library
67 Canal Street
Lyons, New York 14489

Rochester Public Library
115 South Avenue
Rochester, New York 14627

Mr. Robert N. Pinkney
Supervisor of the Town of Ontario
107 Ridge Road West
Ontario, New York 14519

cc w/enclosures & incoming:
Dr. William Seymour
Staff Coordinator
New York State Department of
Commerce
New York State Atomic Energy
Council
99 Washington Street
Albany, New York 12210

Mr. Paul Arbesman
Environmental Protection Agency
26 Federal Plaza
New York, New York 10007

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 7
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Rochester Gas and Electric Corporation (the licensee) dated October 31, 1974, March 11 and April 28, 1975, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Provisional Operating License No. DPR-18 is hereby amended to read as follows:

OFFICE➤						
SURNAME➤						
DATE➤						

"2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 16."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/ RSBoyd

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 16 to
Technical Specifications

Date of Issuance: MAY 14 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A as follows:

Remove Pages	Insert Revised Pages
3.1-24	3.1-24
3.1-29	3.1-29
3.3-2	3.3-2
3.3-4	3.3-4
3.3-12	3.3-12
3.10-3	3.10-3
3.10-4	3.10-4
--	3.10-4a
3.10-8	3.10-8
--	3.10-8a
--	3.10-8b
--	3.10-8c
3.10-9	3.10-9
Table 3.10-3	Table 3.10-3
4.2-1 through 4.2-9	4.2-1 and 4.2-2
4.3-1 through 4.3-3	4.3-1 and 4.3-2

3.1.5 Leakage

Specification:

3.1.5.1 An investigation to determine the location of the leakage from the primary coolant system shall be initiated within 4 hours of a significant increase in leakage. The reactor shall be placed in a hot shutdown condition or the leaking sections isolated within 24 hours after detecting the increase in leakage if:

a. Any leakage from the reactor coolant system pressure boundary is known to be through a pipe, vessel, or valve body, or

b. The known leakage source, other than the above, is greater than 10 gpm, or

c. The leakage source is unidentified and the total unidentified leakage is greater than 1 gpm.

3.1.5.2 Steam generator tube leakage in one steam generator sha
16 not exceed 0.1 gpm when averaged over 24 hours. If thi
limit is exceeded, the reactor shall be shut down withi
8 hours, and an inspection shall be performed. This inspection shall be in accordance with the requirements of Technical Specification 4.2, if more than six months have elapsed since the last steam generator inspection.

3.1.5.3 Two primary coolant boundary leak detection systems of different principles, including one system sensitive to radioactivity, shall be in operation when the reactor is being operated above 5% power. A system sensitive to annual average allowed by 10 CFR Part 20.

Should a postulated transient or accident occur (such as a rod ejection or steam line break accident) then, if the primary to secondary leak rate is limited to 0.1 gpm per steam generator, the site boundary dose would be maintained well within the guidelines and all steam generator tubes would maintain their integrity .

Continuous operability of two systems of diverse principles is desired to assure some surveillance of coolant leakage. However, due to the redundancy of systems designed to monitor degradation of the reactor coolant pressure boundary, provisions for short term degradation of one system or long term substitution of a system do not materially alter the degree of safety.

Reference:

(1) FSAR Section 11.2.3, 14.2.4

a. T refueling water tank conta not less than 230,000 gallons of water, with a boron concentration of at least 2000 ppm.

b. Each accumulator is pressurized to at least 700 psig with an indicator level of at least 50% and a maximum of 82% with a boron concentration of at least 1800 ppm. Neither accumulator may be isolated. 16

c. Three safety injection pumps are operable.

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

g. A.C. Power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D, accumulator injection valves 841 and 865, and refueling water storage tank delivery valves 856. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. As soon as appropriate modifications are complete, D.C. control power shall be removed from refueling water storage tank delivery valves 896A and B with the valves open. In the meantime, single failure protection for valves 896A and B will be provided by locking out A.C. power, remote from the control room, with operating personnel assigned specifically to restore A.C. power when the valves are required to function in the event of a loss-of-coolant accident. 16

h. Revisions to procedures for post-LOCA long term cooling as described in letters to the Nuclear Regulatory Commission from Rochester Gas and Electric Corporation dated April 1, 1975, April 30, 1975, and May 13, 1975, shall be implemented prior to reactor startup following the shutdown of March 10, 1975.

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.1.1 within the

(ii) The two reactor coolant drain tank pumps shall be tested and their operability demonstrated prior to initiating repairs of the inoperable residual heat removal pump.

d. One residual heat exchanger may be out of service for a period of no more than 24 hours.

e. Any valve required for the functioning of the safety injection or residual heat removal systems may be inoperable provided repairs are completed within 12 hours. Prior to initiating repairs, all valves in the systems that provide the duplicate function shall be tested to demonstrate operability.

f. Power may be restored to any valve referenced in 3.3.1.1 g for the purposes of valve testing providing no more than one such valve has power restored and provided testing is completed and power removed within 12 hours. 16

3.3.2 Containment Cooling and Iodine Removal

3.3.2.1 The reactor shall not be made critical except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 4500 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. At least two containment spray pumps are operable.

c. At least three fan cooler units are operable.

until repairs were effected.[6][7]

The facility has four service water pumps. Only one is needed during the injection phase, and two are required during the recirculation phase of a postulated loss-of-coolant accident.[8]

The limits for the accumulator pressure and volume assure the required amount of water injection during an accident, and are based on values used for the accident analyses. The indicated level of 50% corresponds to 1108 cubic feet of water in the accumulator and the indicated level of 82% corresponds to 1134 cubic feet. 16

References

(1) FSAR Section 9.3

(2) FSAR Section 6.2

(3) FSAR Section 6.3

(4) FSAR Section 14.3.5

(5) FSAR Section 1.2

(6) FSAR Section 9.3

(7) FSAR Section 14.3

(8) FSAR Section 9.4

average power tilt ratio shall be determined once a day by at least one of the following means:

a. Movable detectors

b. Core-exit thermocouples

3.10.2.2 Power distribution limits are expressed as hot channel factors. At all times, except during low power physics tests the hot channel factors must meet the following limits:

$F_Q(Z) = (2.32/P)*K(Z)$ for $P \geq .5$

$F_Q(Z) = 4.64*K(Z)$ for $P \leq .5$

$F_{\Delta H}^N = 2.22 - .56P$ for $P \geq .75$

$F_{\Delta H}^N = 1.80$ for $P \leq .75$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given by Figure 3.10-3, and Z is the height in the core. The measured F_Q^N shall be increased by three percent to yield F_Q. If the measured F_Q or $F_{\Delta H}^N$ exceeds the limiting value, with due allowance for measurement error, the maximum allowable reactor power level and the Nuclear Overpower Trip set point shall be reduced one percent for each percent which $F_{\Delta H}^N$ or F_Q exceeds the limiting value, whichever is more restrictive. If the hot channel factors cannot be reduced below the limiting values within one day, the Overpower ΔT trip setpoint and the Overtemperature ΔT trip setpoint shall be similarly reduced.

3.10.2.3 Except for physics tests, if the quadrant to average power tilt ratio, exceeds 1.02 but is less than 1.12 or if a part-length or full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors, and apply Specification 3.10.2.2, or

c. Limit power to 75% of rated power.

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip set point is reduced by 50%.

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by interpolation using the most recent measured value and the predicted value at the end of the cycle life. 16

3.10.2.8 Except during physics tests, control rod exercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within ± 5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

3.10.2.9 Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

3.10.2.10 Except during physics tests, control rod exercises, or exc e calibration, at a pow level less than or equal to 90 percent of rated power:

a. The indicated axial flux difference may deviate from its ± 5% target band for a maximum of one hour (cumulative) in any 24 hour period, however, the flux difference shall not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power.

b. If Specification 3.10.2.10a is violated, then the reactor power shall be immediately reduced to no greater than 50% power. 16

c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.11 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period. One half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90% of rated power.

3.10.2.12 When the reactor is critical and thermal power is less than or equal to 90% of rated power, an alarm is provided to indicate when the axial flux difference has been outside the target band for more than one hour (cumulative) out of any 24 hour period. In addition, when thermal power is greater than 90% of rated power, an alarm is provided to indicate when the axial flux difference is outside the target band. If either alarm is out of service, the flux difference shall be logged hourly for the first 24 hours the alarm is out of service and half-hourly thereafter.

enforceabl…mit below which design dist…ution is not exceeded. In the event that an LVDT is not in service, the effects of a mal-positioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors.

"The two hours in 3.10.2.3 are acceptable since complete rod misalignment (Part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power to 75% will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-3 has been determined from extensive analyses considering operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope demonstrate compliance with the Final Acceptance Criteria limit for Emergency Core Cooling Systems.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map with the movable incore detector flux mapping system. 16

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading pattern. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomolies which might, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

2. Control rod banks are sequenced with overlapping banks as described in Specification 3.10.

3. The full length and part length control bank insertion limits are not violated.

4. Axial power distribution limits which are given 16
in terms of flux difference limits and control bank insertion limits are observed. Flux difference is $q_T - q_B$ as defined in Specification 2.3.1.2d.

The permitted relaxation in $F^N_{\Delta H}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In specification 3.10 F_Q is arbitrarily limited for $P<0.5$ (except for low power physics tests).

The limits on axial power distribution referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies primarily with burnup.

The technical specifications on power distribution assure that the F_Q upper bound envelope of 2.32 times Figure 3.10-3 is not exceeded and xenon distributions are not developed which, at a later time, could cause greater local power peaking even though the flux difference is then within the limits.

The target (or reference) value of flux difference
is determined as follows. At any time that
equilibrium xenon conditions have been established,
the indicated flux difference is noted with part
length rods withdrawn from the core and with
control Bank D more than 190 steps withdrawn.
This value, divided by the fraction of full power
at which the core was operating is the full power
value of the target flux difference. Values for all
other core power levels are obtained by multiplying
the full power value by the fractional power. Since
the indicated equilibrium value was noted, no allow-
ances for excore detector error are necessary and
indicated deviation of ± 5 percent ΔI is permitted
from the indicated reference value. During periods
where extensive load following is required; it may be 16
impossible to establish the required core conditions
for measuring the target flux difference every month.
For this reason, two methods are permissible for
updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power.

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is 16
required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. The is accomplished, without part length rods, by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

The effect of exceeding the flux difference band at or below half power is approximately half as great as it would be at 90% of rated power, where the effect of deviation has been evaluated.

The reason for requiring hourly logging is to provide continued surveillance of the flux difference if the normal alarm functions are out of service. It is intended that this surveillance would be temporary until the alarm functions are restored.

"The quadrant power tilt of 1.02 at which remedial action is required has been set so as to provide DNB and linear heat generation rate (kilowatts/foot) protection in radial power tilts. Analyses have shown that the ratio of increase in $F_{\Delta H}^N$ to increase in quadrant power tilt is less than or equal to 2 to 1. In addition, comprehensive dropped and static ejected rod testing performed during the initial startup program demonstrated that this ratio was less than 1.5 to 1. For conservatism, the 2 to 1 ratio is used.

"The uncertainty factor included during core nuclear design is 1.10 for both F_q^N and $F_{\Delta H}^N$. Therefore, the limiting tilt has been set as 1.02. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the tilt reading and/or to determine and correct the cause of the tilt. Should this action verify a tilt in excess of 1.02 which remains uncorrected, the margin for uncertainty in F_q^N and $F_{\Delta H}^N$ is reinstated by reducing the power by 2% for each percent of tilt above 1.0, in accordance with the 2 to 1 ratio above, or as required by the restriction on peaking factors.

"If instead of determining the hot channel factors, the operator decides to reduce power, the specified 75% power maintains the design margin to core safety limits for up to a 1.12 power tilt, using the 2 to 1 ratio. Reducing the overpower trip set point ensures that the protection system basis is maintained for sustained plant operation. A tilt ratio of 1.12 or more is indicative of a serious performance anomaly and a plant shutdown is prudent." 16

The specified rod drop time is consistent with safety analyses that have been performed.(1)

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one except during physics testing, in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

The reactivity worth limit for an inoperable control rod is consistent with the value found tolerable in the analysis of the hypothetical rod ejection accident.(3) The initial-core physics testing showed the maximum worth to be less than 0.365% when the controlling Group D was more than 60% withdrawn, whereas larger worths were possible with the controlling bank fully inserted.(4) Consequently,

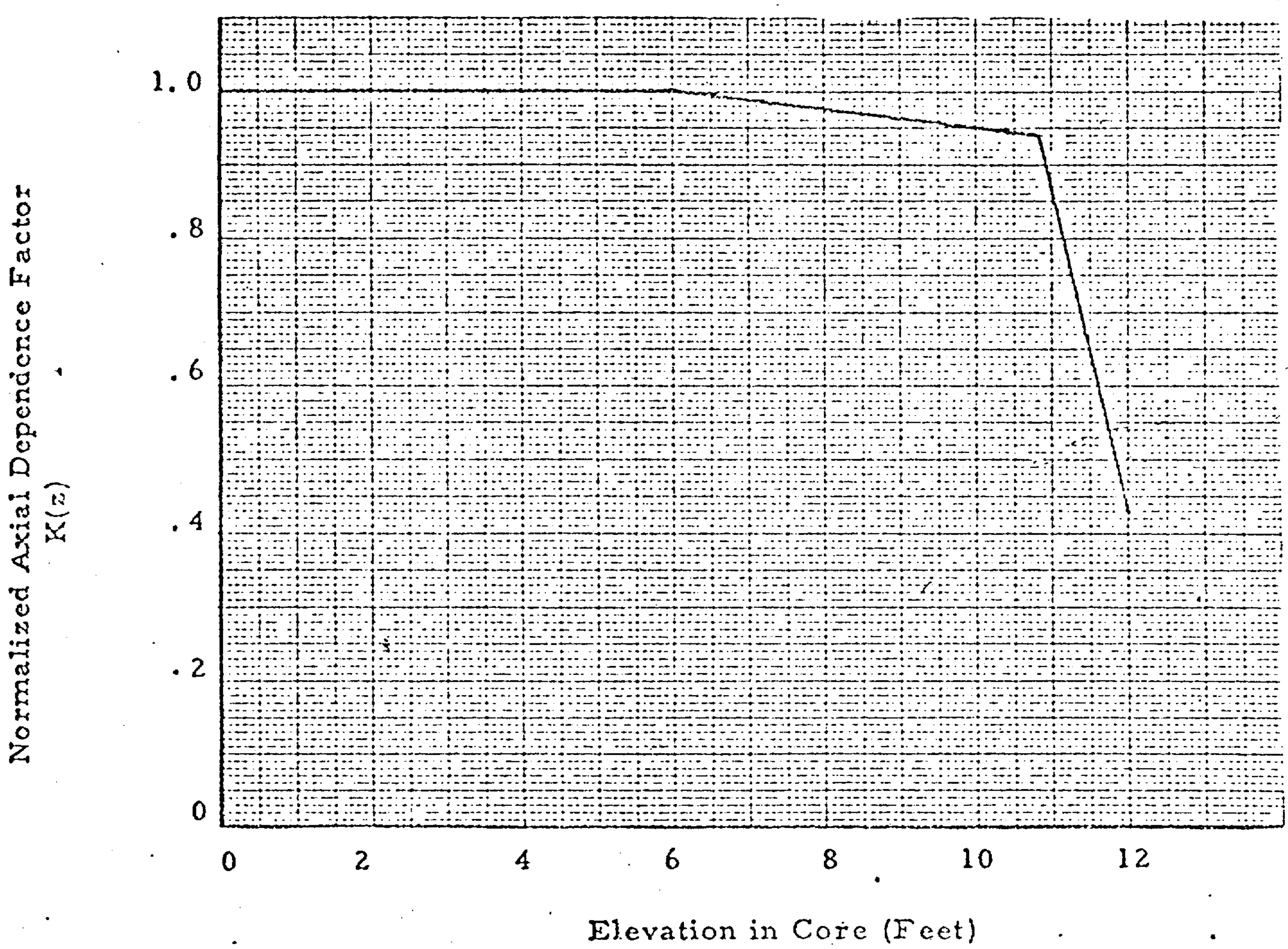

Figure 3.10-3 Normalized Axial Dependence Factor for F_Q versus Elevation

16

3/75

4.2 Inservice Inspection

Applicability

Applies to the inservice inspection of Quality Groups A, B and C Components, High Energy Piping Outside of Containment and Steam Generator Tubes.

Objectives

To provide assurance of the continuing structural integrity of the structures, components and systems.

Specification 16

4.2.1 The inservice inspection program shall be in accordance with Attachment A to the Rochester Gas and Electric Corporation Application for Amendment, dated October 31, 1974 except for the following:

4.2.1.1. All statements regarding "hydrostatic pressure test" in Section 4.2.8 shall not be applicable.

4.2.1.2. In Sections 4.2.6.1, 4.2.6.4, and 4.2.6.5, the term "standard engineering criteria" shall be interpreted to mean "QC 1003 of the Quality Control Manual."

4.2.1.3. Section 4.2.7.4 is changed to read as follows: Steam generator tubes that have defects equal to or greater than 40 percent through-wall as indicated by the eddy current method or an equivalent method shall be repaired by using the explosive tube plugging technique or an equivalent method.

4.2.2 Specification 4.2.1 shall be effective until September 1, 1976.

4.2.3 Prior to March 1, 1976, the licensee shall submit an inservice inspection program that shall meet the requirements, except for design and access provisions and preservice examination requirements, set forth in editions of Section XI of the ASME B&PV Code and Addenda through Summer 1975.

Basis:

This inservice inspection program conforms as far as practicable to the 16
requirements of Section XI of the ASME Boiler and Pressure Vessel Code for the remainder of the present one-third of the inspection interval. The licensee will periodically update this program where practicable to comply with later Section XI requirements with due consideration to physical access. This compliance will constitute an acceptable basis for satisfying the requirements of General Design Criterion 32, Appendix A of 10-CFR Part 50.

4.3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

Applicability:

Applies to the tests of the metallurgical specimens taken from the reactor beltline region.

Objectives:

To provide data for the determination of the fracture toughness of the reactor vessel.

Specification:

4.3.1 The reactor vessel material surveillance testing program is designed to meet the requirements of Appendix H to 10CFR Part 50. This program consists of the metallurgical specimens receiving the following test: tensile, charpy impact and the WOL test. These test of the Radiation Capsule Specimens shall be performed as follows: 16

Capsule	Time Tested
V	End of 1st core cycle
R	End of 3rd core cycle
S	10 years, at nearest refueling
N	20 years, at nearest refueling
T	30 years, at nearest refueling
P	Standby

4.3.2 The report of the Reactor Vessel Material Surveillance shall be written as a Summary Technical Report as required by Appendix H to 10CFR Part 50.

Basis:

This material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of the reactor resulting from exposure to neutron irradiation and the thermal environment. The test data obtained from this program will be used to determine the conditions under which the reactor vessel can be operated with adequate margins of safety against fracture throughout its service life. 16

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-18

GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-18. These changes would authorize the Rochester Gas and Electric Company (the licensee) to operate the Ginna Nuclear Power Plant (located in Wayne County, New York) with changes to the limiting conditions for operation resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). This change is being made in conjunction with a reactor refueling for core cycle 5.

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-18, Ginna Nuclear Power Plant, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for the particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for Ginna Nuclear Power Plant issued in December 1973. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the

Lyons Public Library, 67 Canal Street, Lyons, New York and at the Rochester Public Library, 115 South Avenue, Rochester, New York.

Dated at Rockville, Maryland this 30th day of April 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Wm. H. Regan, Jr., Chief
Environmental Projects Branch 4
Division of Reactor Licensing

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 7 TO DPR-18

CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS

ROCHESTER GAS AND ELECTRIC COMPANY

GINNA NUCLEAR POWER PLANT, UNIT 1

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated March 11, 1975, Rochester Gas and Electric Co. submitted proposed changes to the Technical Specifications Appendix A to Licence DPR-18. The proposed changes resulted from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) in conjunction with a reactor refueling for core cycle 5. Supplemental information relating to the ECCS evaluation has been supplied by Rochester Gas and Electric Co. in their two letters of April 1, 1975. The staff has reviewed this matter and the conclusions are set forth below.

Rochester Gas and Electric Co. is presently licensed to operate the Ginna Nuclear Power Plant, Unit 1, located in the State of New York, Wayne County, at power levels up to 1,300 megawatt thermal (MWt). The proposed change to incorporate the ECCS Acceptance Criteria in conjunction with the core refueling does not result in an increase or decrease in power levels of the unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statement (FES) for Ginna Nuclear Power Plant, Unit 1, Docket No. 50-244, dated December 1973.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for Ginna Nuclear Power Plant, Unit 1, Docket No. 50-244, dated December 1973 can be predicted for the proposed action. The Commission's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and releases rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Ginna Nuclear Power Plant, Unit 1. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

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

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 7 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

(CHANGE NO. 16 TO TECHNICAL SPECIFICATIONS)

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

INTRODUCTION

On December 27, 1974, the[1] Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR §50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". One of the requirements of the Order was that the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, §50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Rochester Gas and Electric Corporation (the licensee) has submitted an ECCS reevaluation and related Technical Specifications. The reevaluation and Technical Specifications, which are applicable to the Ginna reactor for the refueled core (Cycle 5), were submitted in a letter dated March 11, 1975.

By letter dated October 31, 1974, the licensee had also proposed Technical Specification changes to incorporate an updated inservice inspection program for the primary coolant system, steam generators, and other safety related components. Technical Specification changes dealing with inservice inspection have been included in this amendment so that the requirements of the updated inservice inspection program will be applicable for the new operating cycle.

EVALUATION

I. ECCS REANALYSIS

The background of the staff review of the Westinghouse ECCS models and their application to R. E. Ginna Nuclear Power Plant is described in the staff SER for this facility dated December 27, 1974 (the December 27, 1974 SER) issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974(2) and the Supplement to the Status Report of November 1974(3) which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier Westinghouse evaluation model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Ginna ECCS evaluation which is covered by this safety evaluation properly conforms to the accepted model. The March 11, 1975 submittal contained: (1) analyses of sufficient break sizes and locations to verify that the worst break condition had been considered and (2) documentation, by reference to submitted Westinghouse Topical Reports, of the ECCS model modifications described in our December 27, 1974 SER.

The analyses submitted March 11, 1974, identified the worst break size as the 0.4 double-ended cold leg guillotine with a calculated peak clad temperature of 1898°F, well below the acceptable limit of 2200°F as specified in 10 CFR §50.46(b). In addition, the calculated maximum local metal/water reaction of 2.45% and total core wide metal/water reaction of less than 0.3% were well below the allowable limits of 17% and 1%, respectively.

The licensee requested on March 27, 1975, authorization to revise overpower ΔT and overtemperature ΔT set points to permit operation at 2250 psia and a full power T average of 573.5°F. This request is still under review. The analysis upon which the ECCS performance calculations are based assumes the higher primary system pressure of 2250 psia. Use of a higher primary system pressure in ECCS analysis is conservative with respect to resulting peak clad temperature, clad oxidation, and hydrogen generation and conforms to the requirements of 10 CFR §50.46.

The ECCS evaluation computations are based on the use of a recent Westinghouse fuel densification model, WCAP-8219, which is referenced in the approved Westinghouse ECCS cooling performance evaluation model. The present safety settings for the Ginna facility on the combination of thermal power coolant temperature and coolant pressure are based on a somewhat older densification model (WCAP-8058) which results in a lower combination of thermal power, coolant temperature, and coolant pressure than would result from application of the newer model. The licensee has requested that these safety settings be revised to permit the higher combination of thermal power, coolant temperature, and coolant pressure based on the use of the newer model (40 FR 16249, April 10, 1975). This request is still pending. The submitted LOCA analyses were performed at initial conditions (pressure, temperature, and thermal power) corresponding to the proposed safety limits. If the LOCA analyses were performed at the presently authorized lower operating conditions (pressure, temperature, and thermal power) corresponding to the existing Technical Specifications, lower peak clad temperatures would have resulted (See Appendix B to Rochester Gas and Electric Corporation letter to NRC dated September 6, 1974). When the proposed higher safety limit settings on combined thermal power, coolant temperature, and coolant pressure are authorized those settings will correspond to the seatings assumed for the ECCS coolant performance evaluation. In the interim, the assumptions used are conservative.

Our review of plant-specific assumptions regarding the Ginna analysis addressed the areas of minimum containment pressure, long term core cooling with respect to potential boron precipitation concerns, and the single failure criterion.

A. ECCS Containment Pressure Evaluation

The ECCS containment pressure calculations for the Ginna Plant were done using the Westinghouse ECCS evaluation model. We reviewed Westinghouse's model and published a Status Report on October 15, 1974[2], which was amended November 13, 1974[3]. We concluded that Westinghouse's containment pressure model was acceptable for ECCS evaluation and required that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for the Ginna Plant by letter dated November 25, 1974[4]. Rochester Gas and Electric has reevaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for ECCS analysis. This evaluation was based on measurements within the containment and from as-built drawings to which additional margin was added. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Ginna is conservative and therefore the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

B. Single Failure Criterion

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred. The worst single failure which would minimize the ECC available to cool the core and provide maximum containment cooling was identified by Westinghouse as the loss of a low pressure ECCS pump. We concluded in Reference 2 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

A review of the Ginna piping and instrumentation diagrams indicated that the spurious actuation of specific motor operated valves could affect the appropriate single failure assumptions. We identified the following motor operated valves which did not satisfy the single failure criterion.

MOV #	Location
841, 865	Accumulator Isolation Valves
896A, 896B	S.I. Pump Suction from RWST
856	RHR Pump Suction from RWST
878B, 878D	S.I. Pump Discharge to Cold Legs
878A, 878C	S.I. Pump Discharge to Hot Legs

The licensee reviewed the consequences of these spurious failures and proposed the following actions:

1. AC power to be removed from MOV 841, 865, 856, and 878A, B, C, and D. All valves will be in their open position with the exception of 878A and C, which will be closed in accordance with Technical Specification 3.3.1.1.g.

2. MOV 896A and B, which are normally open valves during the injection phase of a LOCA, are required to be closed during the recirculation phase in the event that high head recirculation is required. In order to provide for operator action from the control room during the switchover to recirculation, the licensee has proposed to remove D.C. control

power from these valves in the control room. This would reduce the probability of spurious valve actuation by necessitating two shorts in the three phase A.C. power supply to cause a spurious valve action.

We had previously concluded that locking out of A.C. power to motor operated valves was an acceptable procedure to design against a single failure that could cause an undesirable component action and result in a loss of capability to perform an intended safety function. Branch Technical Position EICSB 18, in Appendix 7A of the Standard Review Plans is to be applied to each of the "active", manually controlled electrically operated valves that is required to operate during various safety system operational sequences.

For MOV 896A and B, we have concluded that the licensee's proposal to remove D.C. control power by the installation of a suitable switch to interrupt the D.C. control circuit in the main control room is an acceptable modification. Although the proposed modification does not eliminate all possible causes for spurious valve actuation, it does significantly reduce the potential problem. This amendment changes the Technical Specifications to require the removal of D.C. power from MOV 896A and B as soon as appropriate modifications are complete. In the meantime, single failure protection for these valves will be provided by locking out A.C. power, remote from the control room, with operating personnel assigned specifically in the auxiliary building to restore A.C. power when the valves are required to function in the event of a loss of coolant accident.

C. Boric Acid Build-up During Long Term, Post-LOCA Core Cooling

The licensee submitted the Ginna emergency operating procedures proposed for the long term post-LOCA core cooling period and indicated that these procedures would prevent excessive concentration of boron in the reactor vessel. The procedures were supported by a Westinghouse analyses[6]. We have reviewed the analyses and proposed procedures. We believe that the analyses are not sufficiently complete to justify the licensee's emergency porcedures. They do demonstrate that the existing ECCS system can be operated in a manner that will prevent excessive boric acid concentration from occurring, provided certain of the proposed procedures are changed. We have required these changes on an interim basis until such time as the licensee has completed further analysis, and we have accordingly reviewed the analyses and modified the Technical Specifications. The procedural changes we have required at this time provide additional margin between the boron concentration and the solubility limit at the time of switch over from cold leg injection to the long-term recirculation mode. Specifically, the licensee has committed[6] and the Technical Specifications provide for the modification of the long-term core cooling procedures to effect switchover occurring at 20 hours of cold leg injection instead of at 24 hours. Hot and cold leg injection would be provided by the low head and safety injection pumps, respectively. The staff has found this procedure to be acceptable.

We have reviewed the ECCS equipment required for the implementation of these procedures and concluded that the presently available equipment would satisfy the applicable General Design Criterion except for specific motor operated valves, located within the containment, which would be submerged at some time during the recirculation mode of operation. These valves may not function when required or may be inadvertently actuated. The licensee will be required to modify the plant design to preclude the failure of these valves and to submit a detailed description of the proposed long-term modifications for staff review and acceptability within 30 days. Until such time as the required modifications are completed, we have changed the Technical Specifications in accordance with the licensee's letter of May 13, 1975, to implement the following interim procedures:

1. MOV 852 A and B, which are normally closed and receive a safety injection signal (SIS) to open, will have A.C. power removed from the motor operators after the valves have been verified to be opened following SIS and before flooding occurs by an operator who will be stationed in the auxiliary building specifically to remove A.C. power at the proper time. An operator to perform this function shall be provided at all times during plant operation.

2. MOV 700, 701, and 721, which are normally closed, do not receive a SIS, and are required only during normal shutdown procedures, will have A.C. power removed whenever the reactor is critical, to preclude their possible inadvertent actuation.

3. The following motor operated valves, which may be flooded, will have A.C. power rocked out during plant operation, as noted in Section B, above:

MOV #878 A, B, C and D
MOV #865
MOV #841

The implementation of these interim procedures will assure that the ECCS equipment required for long-term recirculation meet the applicable General Design Criteria for the period during which the required modifications are being made.

D. Nuclear Design

The core loading for Cycle 5 will include 36 helium prepressurized fresh assemblies, 20 of 3.14 w/o U-235 enrichment and 16 of 3.10 w/o U-235 enrichment. These assemblies have been fabricated by Westinghouse Electric Corporation. Four fuel demonstration assemblies which were present in the core during the previous cycle also are included in the load, two fabricated by Babcock & Wilcox Company, and two by Exxon Nuclear Company. These assemblies are compatible with and substitutable for the Westinghouse fuel in the applicable region.

Calculated core kinetic characteristics for Cycle 5 fall within the range of values assumed for the FSAR analysis. Previous analyses for the accidents affected by these nuclear parameters are therefore applicable and acceptable for Cycle 5. To assure that peak linear heat generation rates consistent with the ECCS reanalysis are not exceeded, the Technical Specifications include: (1) limits on flux difference, (2) flux peaking augmentation factors, (3) power distribution limits, and (4) quadrant power tilt limits.

E. Summary

Our evaluation supports the conclusion that: (1) the evaluation has been performed wholly in conformance with the requirements of Appendix K to 10 CFR §50.46 and (2) ECCS cooling performance for Ginna will conform to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR §50.46(b). In addition, we have concluded that:

1. The single failure criteria will be satisfied.

2. Adequate systems and procedures exist to provide reasonable assurance that boron precipitation will not occur within the reactor vessel.

II. INSERVICE INSPECTION PROGRAM

A. Updated Program

This amendment changes the Technical Specifications to incorporate an updated inservice inspection program for Class A, B, and C components (as defined by Attachment A of the licensee's submittal of October 31, 1974) that will be in effect until August 1976, the end of the current one-third of the ten year inspection interval. It also requires that by March 1, 1976, the licensee submit another updating of the inservice inspection program to be effective for the next one-third of the ten year inspection interval. Our review indicates that the program, as proposed, meets insofar as practicable Section XI of the ASME Boiler and Pressure Vessel Code, and is acceptable.

B. Augmented Inservice Inspection for High Energy Piping Outside of Containment

The augmented inservice inspection program for high energy piping outside of containment proposed by the licensee consists of radiographic examination of all welds at the design basis break locations in the main steam and feedwater lines and at other locations where a failure would result in unacceptable consequences. The examination techniques, procedures, and inspection intervals are based on the requirements for Class 2 components of Section XI of the ASME Boiler and Pressure Vessel Code. The program is based on ten-year inspection intervals with the first interval running from 1973 to 1982.

During each third of the first inspection interval, the program provides for examination of all welds at design basis break locations and one-third of all the welds at locations where a weld failure would result in unacceptable consequences. During the succeeding ten-year intervals, the program provides for examination of one-third of the welds at design basis break locations and one-third of the welds at locations where a weld failure would result in unacceptable consequences during each one-third of the interval. This program is designed to detect flaws capable of causing pipe failure and the frequency of reinspections is designed to detect any change in condition in advance of a potential failure. We conclude that this augmented inspection program is a prudent measure to ensure a very low probability of any break in the main steam and feedwater lines. The requirement for this program is incorporated into the Technical Specifications by this amendment.

C. Steam Generator Tube Inspection

The proposed inservice inspection program provides requirements for steam generator tube inspections in accordance with Regulatory Guide 1.83. We have reviewed the report entitled "Steam Generator Tube Inspection R. E. Ginna Nuclear Plant" dated December 17, 1974, which gives results of the November 1974 inservice inspection and describes plans to change from a phosphate treatment to an All Volatile Treatment (AVT) for the secondary water chemistry upon return to operation in November 1974. We agree with the licensee that the November inspection indicated a reduced rate of tube thinning during a six months operating period (from April 1974 to November 1974) with strict control of the secondary water chemistry, i.e., controlling the Na/PO_4 ratio between 2.3 and 2.6.

We have also reviewed the steam generator tube inspection report submitted by the licensee on April 8, 1975, including Attachment B describing the metallurgical examination of two tubes - one with 99% eddy current (EC) indication and one with 35% EC indication - and the proposed changes to Technical Specification 3.1.5.2. We are in essential agreement with the licensee's conclusion that no additional wastage has occurred since its conversion of the secondary water chemistry to an AVT in November 1974. The basis for this conclusion is that the metallurgical examination of the tube with 35% EC indication showed no additional wastage since the last November inspection. Although the March 1975 inspection results indicate an average of 3% increase in the number of tubes with EC indications in the range of 25% to 49% we agree with the licensee that the increase can be attributed to the improved EC sensitivity. The same explanation can also be applied to the increase in the number of tubes with 20% to 24% EC indications.

The water chemistry data, submitted by the licensee, for the four months operating period indicate that AVT conditions have been stabilized. Therefore, further tube wall thinning by wastage corrosion is not expected to occur during the subsequent full power operation.

It is our opinion that the licensee's proposed tube sheet lancing program, in addition to blowdown during startup, will lower the residual phosphate concentration within the sludge layer and will considerably reduce the probability of further caustic stress corrosion cracking. It is our understanding that the Ginna plant during the May 1975 startup intends to follow staged operation similar to the November 1974 Startup, which should further reduce the residual phosphate concentration.

The results of EC examination of the Ginna Plant steam generator reported by letters of April 8 and April 24, 1975, and the detailed examination of two tubes extracted by the licensee indicate that the cracking caused by caustic stress corrosion following changeover from phosphate to all volatile control is identical to cracking seen at other facilities and that it is a random phenomenon which does not occur preferentially in the thinned regions due to wastage. However, since a possibility exists that cracking may occur in previously thinned regions, the licensee has proposed a revised plugging limit of 40%. This plugging limit is more conservative than the 50% plugging limit at which tubes were plugged earlier at the Ginna facility. This additional conservatism is intended to compensate for a possible reduction in strength caused by caustic stress corrosion cracks occurring in previously thinned regions. The licensee has submitted initial analytical and experimental data in support of the revised plugging limit. We have reviewed this data and have performed independent calculations to examine the adequacy of the proposed plugging criteria.

The combined effect of wall thinning and through-wall cracking on tube strength was determined experimentally with artificially defected tubing. The experimental data indicates that tubes with 40% local wall thinning and a superimposed 0.4 inch long through-wall crack require a pressure of over 3000 psi before bulging occurs. This pressure is greater than the maximum theoretical pressure differential that could be imposed during the most severe postulated event, a main steam line break, which is under 2500 psi. Westinghouse has reported, based on experiment, that leak rates for 0.4 inch long cracks in 0.05 inch wall tubes are 0.1 gpm and leak rates are substantially higher for longer cracks. Similar cracks in thinner tubes are expected to produce more leakage. It has been experimentally observed that leak rates increase in an orderly manner near the pressures at which bulging occurs, even for relatively short cracks. The leakage limit of 0.1 gpm poses no monitoring problems. The staff has also independently determined the structural integrity of tubes with a wall thinning of 40% (0.029-inch wall) in the absence of through-wall cracks.

The maximum stress calculated for the thinned tube, under the maximum operating pressure differential of 1500 psi, was 20.1 Ksi which compares with the minimum yield strength of the tube material (Inconel-ASME SB-163) of 27.9 Ksi at the 600^{o}F operating temperature. Based on an ultimate strength of 75 Ksi at 600^{o}F, the burst pressure of a thinned tube has been conservatively calculated to be a factor of 4.3 over the maximum differential pressure to be contained by the Ginna steam generator tubes during normal operation. The conservatively calculated burst pressure is over twice the full operating pressure of the primary system at Ginna; thus, no rupture of the tubes would result if a steam line break occurred. Even with conservative models, acceptable margins of safety are preserved for the wasted tubes under LOCA plus SSE loads.

It is therefore the conclusion that:

1. Tubes with 40% eddy current indication retain adequate strength to withstand the most severe accident conditions of a main steam line break as well as the combined effects of a LOCA plus SSE.

2. During normal operation the factor of safety against burst is greater than three and the maximum stress levels for the limiting case do not exceed the elastic range of the tube materials under normal operating conditions.

3. Monitoring the primary to secondary leakage rate provides a reliable means for detecting small through-wall cracks which may escape detection during EC inspection. A limit of 0.1 gpm per steam generator on the primary to secondary leakage assures that no through-wall crack is longer than 0.4 inch.

In conclusion, we believe that with the proposed changes to the Technical Specifications, specified AVT chemistry, and planned startup procedures, the Ginna steam generators are acceptable for return to full power operation until the next planned refueling outage.

We have reviewed the proposed inservice inspection program with respect to material surveillance and find it to be consistent with the requirements of Appendix H to 10 CFR Part 50, and acceptable.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: MAY 14 1975

REFERENCES

1. "Order for Modification of License" letter sent to Rochester Gas and Electric Corporation from Robert A. Purple, December 27, 1974.

2. "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.

3. "Supplement to the 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", November 13, 1974.

4. "Additional Information Regarding the Loss-of-Coolant Accident Analysis, R. E. Ginna Nuclear Power Plant, Unit 1", to E. G. Case from K. W. Amish, November 25, 1974.

5. CLC-NS-309, Letter to T. M. Novak from C. L. Caso, Westinghouse Nuclear Energy Systems, April 1, 1975.

6. "Long Term Cooling, R. E. Ginna Nuclear Power Plant Unit No. 1", Letter to Nuclear Regulatory Commission from Rochester Gas and Electric Corporation, April 30, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 to Provisional Operating License No. DPR-18 issued to Rochester Gas and Electric Corporation which revised Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant located in Wayne County, New York. The amendment is effective as of its date of issuance.

The amendment (1) changes operating limits in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR Section 50.46; (2) terminates restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License, and imposes instead, limitations established in accordance with 10 CFR Section 50.46; (3) incorporates an updated inservice inspection program for safety related components to (a) meet Section XI of the ASME Boiler and Pressure Vessel Code, (b) provide an augmented inservice inspection program for high energy piping outside of containment, and (c) provide requirements for steam generator inspection consistent with Regulatory Guide 1.83; and (4) decreases the maximum permissible steam generator leakage from 1 gpm to 0.1 gpm to avoid operation with significant steam generator tube cracks.

Notice of Proposed issuance of Amendment to Provisional Operating License in connection with items (1) and (2) was published in the Federal Register on March 24, 1975 (40 FR 13051) and in connection with item (3) was published in the FEDERAL REGISTER on March 28, 1975 (40 FR 14125). No request for a hearing or petition for leave to intervene was filed following notice of the proposed actions.

For further details with respect to this action see (1) the applications for amendment dated October 31, 1974, March 11 and April 28, 1975, and Supplements dated April 1 (2 letters), 8, and 30, 1975, and May 13, 1975; (2) Amendment No. 7 to License No. DPR-18, with Change No. 16; (3) the Commission's related Safety Evaluation; and (4) the Commission's Negative Declaration dated April 30, 1975, which is being published concurrently with this notice, and associated Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Lyons Public Library, 67 Canal Street, Lyons, New York 14489 and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627.

A copy of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 14th day of May 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing