October 22, 1982 SECY-82-431

ADJUDICATORY ISSUE

(Notation Vote)

For: The Commission

From: Martin G. Malsch
Deputy General Counsel

Subject: REVIEW OF DIRECTOR'S DECISION DD-82-11 (MATTER OF ROCHESTER GAS AND ELECTRIC CORP.)

Facility: Ginna Nuclear Power Plant

Review Time Expires: November 2, 1982

Purpose: To inform the Commission of a Director's Decision

Ex. 5

Discussion: On March 11, 1982 Ruth Caplan of the Sierra Club requested NRC to impose a set of proposed conditions on the Ginna plant before it resumed operation after the January 25, 1982 steam generator tube rupture accident. In a May 22 decision, the Director of NRR granted in part and denied in part the relief requested based on the set of conditions the staff had already devised for Ginna. DD-82-3, 15 NRC 1348. Subsequently, while that decision was pending Commission review, Ms. Caplan filed a letter dated June 10 listing areas where she believed the staff analysis was faulty (Attachment 1).

Ex. 5

CONTACT:
Mark E. Chopko, OGC
634-1493

142

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 5
FOIA- 92-436

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PDR FOIA
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Ex.5 [_______________], DD-82-3 became final agency action on July 21. On October 8, the Director denied relief, DD-82-11, 16 NRC ___ (Attachment 2).

The June 10 Caplan letter listed five areas for additional staff analysis: thermal shock issues, */ safety valve malfunction, potential for iodine release in excess of NRC requirements, inability to interpret plant conditions to anticipate the tube rupture, and whether the PORV should be "safety grade." [

Ex.5

Ex.5

Recommendation: Ex.5 [

Martin G. Malsch
Deputy General Counsel

Attachments:
1. Caplan 6/10/82 Ltr
2. DD-82-11 (10/8/82)

*/ Those issues were the presence of a flaw at the inlet nozzle to vessel weld, and whether the licensee and staff applied proper assumptions in performing analysis of the beltline weld and B-loop circulation.

SECY NOTE: We have asked for Commission action by c.o.b. November 2, 1982 to be consistent with the present expiration of the review time. OGC believes the issues in the paper are such that the Commission can act promptly, if additional time is needed, please advise SECY.

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. Tuesday, November 2, 1982.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Wednesday, October 27, 1982, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

DISTRIBUTION:
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ATTACHMENT 1

SIERRA CLUB

530 Bush Street San Francisco, California 94108 (415) 981-8634

Reply to: 278 Washington Blvd.
Oswego, New York 13126

JUN 14 A10:35

June 10, 1982

Chairman Nunzio J. Palladino
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Palladino:

On March 11, 1982, the Sierra Club filed a show cause petition with the Office of Nuclear Reactor Regulation requesting that the Ginna operating license be suspended or, in the alternative, permission to restart the reactor be withheld, until critical safety issues were reviewed relating to the January 25th accident. On May 22, the Sierra Club was served with a response to its petition. The response made extensive reference to the "Safety Evaluation Report Related to the Restart of the R.E. Ginna Nuclear Power Plant", NUREG 0916, which was issued the same day and which constituted staff permission for the restart of the reactor.

Although the Sierra Club was not given an opportunity to review the NNR's response prior to restart, we have now completed a preliminary review of the staff response, including NUREG 0916. Our review leads us to conclude that several critical safety issues raised in our petition have not been adequately dealt with by staff and that permission for restart should not have been granted before proper resolution of these issues had occurred. We wish to bring these issues to your attention at this time and to encourage the Commission to exercise its authority under 10 CFR 2.206(c) to review these issues raised in the show cause petition.

A. Thermal shock. The Sierra Club finds staff's discussion of thermal shock consequences to the reactor (Club petition at #13) to be seriously deficient. The Safety Evaluation Report does not discuss reactor material properties and irradiation effects except in the most cursory manner and fails to provide adequate analysis of the B loop circulation during the course of the accident.

1. Properties of vessel, nozzle and welds. In the SER, staff fails to evaluate material supplied by licensee in its April 12th report, "Incident Evaluation, Ginna Steam Generator Tube Failure Incident," and in its April 26th supplement, "Affect of Thermal Transient on Reactor Coolant System." These reports discuss the material properties and irradiation effects of the beltline vessel weld, the reactor vessel nozzle and nozzle weld. We have reviewed these reports and wish to bring to the Commission's attention several deficiencies which we consider to be significant.

a. Inlet nozzle to vessel weld. Licensee analyzes the properties of the vessel nozzle, but fails to make any mention of the fact that "an indication in the inlet nozzle N2B to vessel weld that exceeded Code allowable limits was detected" during the in-service inspection performed February-March, 1979, and that the flaw was found to be 0.9 inches in length. (Source: NUREG 0569, "Evaluation of the Integrity of SEP Reactor Vessels," Appendix G, page 80, emphasis added.) At the same time, licensee takes pains to point out that past in-service inspection of the nozzle corners has shown them "to be free of unacceptable ultrasonic indications." (April 12th report at 6.4-3) Although the licensee discusses critical flaw depths for the nozzle, there is again no mention of the nozzle weld. Given that 0.75" is found to be sufficient for a flaw to initiate at the surface of the nozzle itself and to propagate in length and that a flaw deeper than 1.9" can propagate through the thickness of the nozzle, the Sierra Club finds it surprising that the 0.9" weld flaw is ignored.

b. Beltline weld analysis. NUREG 0569 has determined that the beltline weld is the limiting reactor vessel material(Ibid. at 78). Yet licensee's analysis of the potential impact of the Ginna accident on the beltline weld is not sufficiently conservative. The "no warm prestressing" assumption, used for the perfect mixing case, is dropped when the imperfect mixing case is considered. Licensee asserts that, having used the conservative mixing assumption, they should not also have to add the conservative assumption of "no warm prestressing." They conclude: "For the no mixing case, using the modified Reg. Guide 1.99 trend curve and the warm prestressing principle, no flaw was found to initiate." (April 26th report at 4.1) This leaves the reader wondering whether a flaw would be found to initiate when warm prestressing is not assumed. Staff should have required that this question be answered.

2. Staff analysis of B loop circulation. The thermal shock analysis provided by the Task Force in NUREG 0909 and reiterated with some elaboration in NUREG 0916 at 3.5.2, is not, in our opinion, adequate to support staff's contention that flow reversal in the B loop prevented cold water as measured by the temperature sensor from entering the reactor vessel.

Staff has apparently made no attempt to model the hydrodynamics of the primary loop flow during the period of temperature drop. Such a model must not only account for the mass balance, but also for all relevant dynamics such as buoyant and viscous forces and turbulent mixing. Lacking such a model which integrates the various forces, staff's attempts at explanation of the system dynamics remain unconvincing. For instance, staff suggests that the steam generator is a heat source which causes loss of natural circulation flow in the B-loop, without mentioning any other factors which would effect flow.

Other potentially important dynamics are ignored by staff. For instance, staff fails to discuss the flow consequences of the RCS pressure falling below the S/G B pressure, resulting in reverse flow through the tube rupture during the PORV openings. Nor does staff attempt to analyze the dynamics by which water lost from the B loop through the burst tube and PORV is replaced in the system. The question of stratified flow with some cold safety injection water being drawn into the reactor is certainly not answered by staff's vague reference to use of EPRI data. (NUREG 0916 at 3-15)

Staff asserts that even if cold water had entered the reactor, fracture mechanics analysis indicates that there would be no crack initiation. We are given almost no information about this analysis; however, we are told that the temperature used was that measured by the sensor in the cold leg of the B loop. (Ibid.at 3-15) This is portrayed as a worst case analysis, despite staff's recognition on the previous page that the temperature entering the reactor could be 10° less than the measured temperature.

c. Conclusion. In summary, the Sierra Club finds the presentation in Section 3.5 of NUREG 0916 to be incomplete and unconvincing. Substantial question remains regarding thermal shock to the reactor. The existence of the nozzle weld flaw is never mentioned by licensee or staff. Nor has the fracture mechanics analysis been truly conservative. Given the resulting uncertainty combined with the age of the Ginna reactor (8 EFPY), the prudent course would be to require ultrasonic testing of the reactor vessel and nozzle weld. Such testing should take advantage of newer techniques which use multi-angled probes and time of flight information as recommended by Cottrell (New Scientist 25 March 1982, page 775). Such testing should be required before the Commission allows continued operation of the reactor.

B. Safety valve. The Sierra Club considers staff response regarding the safety significance of the steam generator safety valve malfunction and the lack of any proposed corrective action to be an unacceptable response to the Club petition #11b. We wish to bring this concern to the Commission's attention.

The Task Force, appointed by the Commission, determined that the safety valve opened and closed five times. Staff in NUREG 0916 notes the Task Force findings regarding the malfunction of the valve in the following passage:

> "NUREG 0909 also notes that the valve opened and closed at generally decreasing pressures and discussed a possible reason for the decreasing closing pressures; the possibility of some steam leakage after closing the first time, and water leakage estimated at 100 gpm after the last closing. The NUREG attributed the water leakage to the likelihood of failure to fully reseat after the last closing until 50 minues later when the valve apparently stopped leaking."
> (NUREG 0916 at 6-11)

Despite this release of approximately 5000 gallons of cooling water contaminated via the tube rupture and released directly to the environment, the staff concludes "that the valve behavior was entirely within its design basis," (Ibid at 6-12) and that "The performance of the steam generator safety valve that opened was satisfactory."

(Ibid. at 6-14). The Sierra Club is shocked by staff's conclusions. hen the safety valve leaks or sticks open, there is no way operators can close the valve manually. Nor can a block valve be closed. During a SGTR accident, the safety valve is a direct path for loss of radioactive steam or water to the environment. The potential for exceeding Part 100 release limits during a design basis SGTR accident is discussed in the next section. Given this scenario, staff's conclusion that the safety valve is acceptable does not serve to increase citizen confidence in the nuclear industry's ability to protect public health and safety. We are not reassured by staff's decision to give the licensee 6 months in which to review its procedures for a tube rupture with failed SG safety or relief valve. (Ibid. at 4.1.12)

If the safety valve malfunctioned while still meeting the design basis specifications, then the specifications are clearly inadequate. The Ginna reactor should not be allowed to operate without an improved safety valve.

C. Iodine release. Staff recognizes, as a result of the Ginna accident, that "the potential exists for doses [of iodine to be released] exceeding Part 100 Guidelines for a design-basis SGTR accident." (Ibid. at 8-1) As recently as June 25, 1981, staff's analysis of such an accident contained in "Systematic Evaluate Program Evaluation of a Steam Generator Tube Rupture Accident at Ginna" had not considered the possibility of substantial amounts of water and steam being released through the safety valve. The inability of staff to model possible accident parameters accurately in advance of an accident lays open to question the basis on which regulations are promulgated.

While we commend staff's caution in reducing the spiking and equilibrium concentration limits for iodine in the primary coolant, we note that staff is willing to remove these stricter standards if licensee can demonstrate that steam generator flooding will not occur. (Ibid. at 8.1) Yet the steam generator did flood with water when it was not expected to do so. At the very least there should be a "lesson learned" from the Ginna accident that such flooding should be part of a design basis SGTR accident.

We note that staff again avoids dealing with the fact that the safety valve is not designed to handle water, or to be cycled open and closed. Staff suggests that the steam generator PORV is better suited for cycling and so "may be better to use." (Ibid. at 8-3) However, staff concedes earlier in its discussion that the relief valve is also subject to malfunction. They state:

> "Two-phase flow through the relief or safety valves may contribute to valve degradation and possible failures to reseat. This can contribute to the radiological consequences by providing a rpolonged pathway to the environment." (Ibid. at 8.1, emphasis added.)

Thus, simply changing the emergency operator guidelines to ensure that the block valve is not closed incorrectly will not remedy the problem. Staff has approved other changes which relate to termination of the safety injection. We are concerned that these changes may have ramifications for core cooling. We are particularly concerned about the following note to be added after STEP 3.15.3:

> "Termination of SI with suspected voids in the upper RV head is allowed when natural circulation is verified." (Ibid. at 8.1)

The Ginna accident has demonstrated how difficult it can be to verify natural circulation. We find no analysis of the consequences of terminating SI with a vessel void, if operators make an error in verifying natural circulation. Nor do we find any analysis of possible adverse consequences of adding STEP 3.20.3 which requires that operators "Block SI before the faulted S/G drops below 550 psig."

Staff admits that there has been "incomplete evaluation of the effects of changes to operator guidelines," (Ibid.) which is one reason the iodine limits are being lowered. The Sierra Club urges the Commission to reconsider the wisdom of allowing Ginna to restart when operating guidelines have been changed without complete evaluation of the safety repurcussions of these changes.

D. Steam Generator Tubes. In response to concerns raised in Sierra Club's petition at #2 a, b, c and #3 regarding in-service inspection standards and specifications for tube rejection, staff simply renumerates the current standards and RG&E procedures. There is no recognition by staff that the inability to anticipate the January 25th tube burst, despite recurrent problems in wedge area #4 and eddy current indication in April, 1981, for the tube that later burst, should be a warning that the standards are not adequate. The Sierra Club is concerned that staff has avoided dealing with the implications of the tube burst and urges the Commission to review the adequacy of these standards.

E. PORV. The Sierra Club raised the concern that the PORV is not required to be safety grade in its petition at #7 and asked for staff review in light of the Ginna accident and the failure of the PORV. Staff has responded that a generic study is underway.(Denton response of May 22, page 5) The fact that a specific cause has been determined for the Ginna PORV failure in no way obviates the importance of making the PORV safety grade. How many accidents involving a malfunction of the PORV need to take place before the staff determines that these valves need to be upgraded? This question is ripe for Commission consideration.

The points raised in this letter are intended only to highlight our concerns regarding staff's response to our petition and are not an exhaustive discussion of every concern. The Sierra Club is hopeful that the Commission, sharing the safety concerns which we have raised herein, will review our petition on its own motion and will reverse staff's decision to allow restart of the Ginna reactor before critical safety issues have been adequately resolved.

While in this letter we have focused specifically on the implications of the accident for the safe operation of the Ginna reactor, we do wish to note that a number of the issues raised have potentially generic significance. Where generic investigations are not already underway, we hope that the Commission will institute such proceedings so that the "lessons learned" from the Ginna accident will not be lost.

Very truly yours,

Ruth N. Caplan, Chair
National Energy Committee
Sierra Club

cc. Samuel J. Chilk, Secretary to the Commission

ATTACHMENT 2

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

October 8, 1982

'82 OCT 12 A8:10

Docket No. 50-244
DD-82-11

Ms. Ruth Caplan, Chair
Sierra Club National Energy Committee
278 Washington Boulevard
Oswego, New York 13126

Dear Ms. Caplan:

SUBJECT: DIRECTOR'S DECISION UNDER 10 CFR 2.206
(R. E. Ginna Nuclear Power Plant)

This is in response to your letter dated June 10, 1982 to Chairman Nunzio J. Palladino which requests that the Commission exercise its authority under 10 CFR Section 2.206(c) to review the issues which you raised in your petition dated March 11, 1982. In that petition you requested that the operating license for Ginna be suspended, or permission to restart the reactor be withheld, until critical safety issues were reviewed relating to the January 25, 1982 steam generator tube rupture at the Ginna plant. By my decision (No. DD-82-03) dated May 22, 1982, I denied the portion of your March 11, 1982 request relating to suspension of operation (47 FR 24491, June 4, 1982). However, as you know, I granted your request that our review include and consider specific areas detailed in that petition prior to restart of the Ginna plant. The documentation of this review is contained in NUREG-0916.

With respect to your June 10, 1982 request, you were advised by letter dated July 28, 1982 from Mr. Samuel J. Chilk, Secretary of the Commission, that the Commission decided to refer this request to the NRC staff for appropriate action under 10 CFR 2.206.

We have considered your request under the provisions of 10 CFR 2.206 of the Commission's regulations. This office has determined, for the reasons set forth in the enclosed decision, not to issue an order suspending the license for operation of the Ginna Plant.

A copy of this determination will be placed in the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555 and at the Local Public Document Room at the Rochester Public Library, 115 South Avenue, Rochester, New York 14604.

The decision will also be filed with the Secretary of the Commission for its review in accordance with 10 CFR 2.206(c) of the Commission's regulations. As provided for by this regulation, the decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance of the decision unless the Commission, on its own motion, institutes a review of the decision within that time.

A copy of the Notice of Issuance of the Director's Decision, which is being filed with the Office of the Federal Register for publication, is also enclosed.

Sincerely,

HR Denton

Harold Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Director's Decision
2. Notice of Issuance

cc w/incoming:
See next page

Ms. Ruth Caplan

cc
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U. S. Environmental Protection Agency
Region II Office
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26 Federal Plaza
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
HAROLD R. DENTON, DIRECTOR

In the Matter of	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION	)	Docket No. 50-244 (10 CFR 2.206)
	)	
(R. E. Ginna Nuclear Power Plant)	)	

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I.

By a letter dated June 10, 1982, Ms. Ruth Caplan, Chair, Sierra Club National Energy Committee, requested that the Commission exercise its authority under 10 CFR Section 2.206(c) to review the partial denial (DD-82-03) by the Director of Nuclear Reactor Regulation of Ms. Caplan's petition dated March 11, 1982. In the March 11 petition Ms. Caplan requested that the Director of Nuclear Reactor Regulation initiate a review of matters pertaining to the ability of the licensee to safely operate the Ginna plant so as to protect public health and safety in light of the January 25, 1982, steam generator tube rupture (SGTR) event at the Ginna plant. The petitioner further requested that this review be incorporated into the review which was in progress by the staff at that time and that it should include, but need not be limited to, several specific areas discussed in the petition. Pending completion of this review the petitioner requested that the operating license for Ginna be suspended, or in the alternative, restart of the reactor not be permitted.

On May 22, 1982, I denied the portion of Ms. Caplan's request relating to suspension of operation. However, I granted the petitioner's request that the review include and consider specific areas detailed in the petition prior to restart of the Ginna plant. The documentation of this review is contained in the Safety Evaluation Report Related to the Restart of the R. E. Ginna Nuclear Power Plant, NUREG-0916 (May 1982). See Director's Decision, DD-82-03, 15 NRC _____ (May 22, 1982).

On July 21, 1982, the Commission declined to review the partial denial of Ms. Caplan's March 11 petition, but it referred Ms. Caplan's June 10, 1982, letter to the NRC staff for further consideration in accordance with 10 CFR 2.206. I have reviewed the information submitted by Ms. Caplan's June 10, 1982 letter and other information pertinent to the issues addressed therein, as indicated in the following discussion. The significant assertions of her petition are excerpted below.

II.

<u>Petitioner's Assertion and Request</u>

A.1.a <u>Inlet nozzle to vessel weld</u>. Licensee analyzes the properties of the vessel nozzle, but fails to make any mention of the fact that "an indication in the inlet nozzle N2B to vessel <u>weld</u> that exceeded Code allowable limits was detected" during the in-service inspection performed February-March, 1979, and that the flaw was found to be 0.9 inches in length. (Source: NUREG 0569, "Evaluation of the Integrity of SEP Reactor Vessels," Appendix G, page 80, emphasis added.) At the same time, licensee takes pains to point out that past in-servite inspection of the nozzle corners has shown them "to be free of unacceptable ultrasonic indications." (April 12th report at 6.4-3) Although the licensee discusses critical flaw depths for the nozzle, there is again no mention of the nozzle weld. Given that 0.75" is found to be sufficient for a flaw to initiate at the surface of the nozzle itself and to propagate in length and that a flaw deeper than 1.9" can propagate through the thickness of the nozzle, the Sierra Club finds it surprising that the 0.9" weld flaw is ignored.

Response:

The subject ultrasonic (UT) indication was detected in the B recirculation inlet nozzle-to-shell weld during the scheduled 10-year inservice inspection conducted in February 1979. Due to the configuration of the nozzle, scanning with the ASME Code required UT procedure (0° longitudinal wave and 45° and 60° angle beam sheer waves) did not reveal any indications. RG&E also examined the nozzle with a 15° refracted longitudinal wave and a 45° sheer wave in accordance with the methods and techniques described in Appendix I of Section XI of the ASME Code and detected the indication with only the 15° longitudinal wave. Based on the 50-50 DAC (Distance Amplitude Correction) sizing criterion, the reported indication has dimension of 0.93 inches in through-wall depth and 5.27 inches in length which is larger than the code allowable standard specified in Table IWB-3512.1 of the Summer 1974 Addenda to the Section XI Code. However, when the beam spread correction at 50% DAC was employed, which was later reviewed and accepted by the staff, this near mid-thickness indication became a code acceptable flaw. This is the reason why the staff would not have expected this nozzle-to-shell weld indication to be mentioned in the licensee's April 12, 1982 report. This indication is believed to correspond to the entrapped slag observed in the fabrication radiograph and no significant growth existed in this weld based on the 1979 inspection. Furthermore, the pressure-temperature transient experienced during the January 25, 1982 tube rupture event did not result in the pressure-temperature changes exceeding those considered in the Design Transient Specifications. Therefore, reevaluation of this matter is not necessary to ensure the vessel integrity.

The stated critical flaw depth for crack initiation refers to an inside diameter surface crack and was determined to be 0.75", assuming a large LOCA with injection water at 70°F. This assumed trans' is much more severe than the Ginna event. Also, the peak thermal stresses during a cooldown transient are at the cooled surface and the normal procedure is to postulate that the critical flaw is at this surface. The Ginna indication (not necessarily a crack) is deeper within the vessel wall and, hence, would not be subjected to these high thermal stresses. Thus, even if it were a crack as large as 0.93", it would not be expected to initiate. Also, the metal temperature and hence its toughness at this internal location would be higher than at the surface which is another factor that would preclude crack growth. Further, the calculation referred to a postulated flaw in the irradiated beltline weld, whereas the flaw actually found was in the nozzle to shell weld, far from any radiation level that could cause significant reduction in fracture toughness.

Petitioner's Assertion and Request

A.1.b Beltline weld analysis. NUREG 0569 has determined that the beltline weld is the limiting reactor vessel material (Ibid. at 78). Yet licensee's analysis of the potential impact of the Ginna accident on the beltline weld is not sufficiently conservative. The "no warm prestressing" assumption, used for the perfect mixing case, is dropped when the imperfect mixing case is considered. Licensee asserts that, having used the conservative mixing assumption they should not also have to add the conservative assumption of "no warm prestressing."

They conclude: "For the no mixing case, using the modified Reg. Guice 1.99 trend curve and the warm prestressing principle, no flaw was found to initiate." (April 26th report at 4.1) This leaves the reader wondering whether a flaw would be found to initiate when warm prestressing is not assumed. Staff should have required that this question be answered.

A.2 Staff analysis of B loop circulation. The thermal shock analysis provided by the Task Force in NUREG 0909 and reiterated with some elaboration in NUREG 0916 at 3.5.2, is not, in our opinion, adequate to support staff's contention that flow reversal in the B loop prevented cold water as measured by the temperature sensor from entering the reactor vessel.

Staff has apparently made no attempt to model the hydro-dynamics of the primary loop flow during the period of temperature drop. Such a model must not only account for the mass balance, but also for all relevant dynamics such as buoyant and viscous forces and turbulent mixing. Lacking such a model which integrates the various forces, staff's attempts at explanation of the system dynamics remain unconvincing. For instance, staff suggests that the steam generator is a heat source which causes loss of natural circulation flow in the B-loop, without mentioning any other factors which would effect flow.

Other potentially important dynamics are ignored by staff. For instance, staff fails to discuss the flow consequences of the RCS pressure falling below the S/G B pressure, resulting in reverse flow through the tube rupture during the PORV openings. Nor does staff attempt to analyze the dynamics by which water lost from the B loop through the burst tube and PORV is replaced in the system. The question of stratified flow with some cold safety injection water being drawn into the reactor is certainly not answered by staff's vague reference to use of EPRI data. (NUREG 0916 at 3-15)

Staff asserts that even if cold water had entered the reactor, fracture mechanics analysis indicates that there would be no crack initiation. We are given almost no information about this analysis; however, we are told that the temperature used was that measured by the sensor in the cold leg of the B loop. (Ibid. at 3-15) This is portrayed as a worst case analysis, despite staff's recognition on the previous page that the temperature entering the reactor could be 10° less than the measured temperature.

Response:

The staff is currently performing an analysis of the R. E. Ginna steam generator tube rupture event of January 25, 1982. The RETRAN 02[1] computer program is being used to perform this analysis. Results of this analysis are expected to be completed by the end of the year. We believe this analysis will support the conclusions of NUREG-0916 concerning pressurized thermal shock.

In support of the staff findings, the following additional information is provided concerning the analyses performed in NUREG-0916:

1. Temperature History Effect

   Due to the thickness and thermal conductivity of the vessel wall, temperature changes of the coolant at the vessel surface propagate more slowly in the vessel wall. The thermal time constant of the wall is on the order of 30 minutes[2]. An example of the temperature distribution in a vessel wall as a function of time, for the specified thermal transient, is shown in the attached figure. Temperature fluctuations in the water, the period of which is a few minutes or less (for example, less than the vessel wall thermal time constant), have little effect on the temperature distribution in the wall and it is possible to use the average surface temperature curve in fracture

(1) "RETRAN 02, A program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM, May 1982.

(2) The time it takes for the bulk (volume average) wall temperature to reach 63% of its final value due to a step change in temperature at the vessel surface.

mechanics analyses. The Ginna SGTR event falls into this category. The effect of the vessel inner wall heat transfer coefficient is the greatest in the most rapidly changing parts of a transient. Note that for the case illustrated, the metal surface temperature as a function of time can be closely approximated by T(wall) = 550-240 [1 -exp (-0.45*t)], if a vessel inner wall heat transfer coefficient of infinity is used. Our studies to date indicate that the most critical factor with respect to pressurized thermal shock considerations is the final temperature of the water. Although our best judgment at this time is that B loop flow was in the direction of the B steam generator during the time the PORV was stuck open, we have conservatively assumed that the B loop flow was towards the vessel for the entire duration of the transient. In this case, the appropriate thermal characteristic (vessel downcomer coolant temperature versus time) for the Ginna SGTR event is that specified as Case 1, from Figure 2.4 of NUREG-0916, with uncertainties associated with instrument errors and mixing of the cold safety injection water. Case 2, from Figure 2.4 of NUREG-0916 is a conservative lower bound of the B loop coolant temperature designed to emcompass the short duration coolant temperature decrease associated with the open PORV. This lower bound is equivalent to adding a total uncertainty of over 60°F to the Case 1 figure. Thus, the Case 2 temperature characteristic bounds the estimated uncertainties in the downcomer temperature (10°F to 20°F for mixing plus 15°F to 25°F for instrument errors). The conclusion that no crack initiation occurred during the SGTR event, based on the Case 2 fracture mechanics analysis, is, therefore, confirmed.

2. Detailed Fracture Mechanics Analysis

A specific, detailed fracture mechanics analysis[3] was performed by Oak Ridge National Laboratories (ORNL) for the R. E. Ginna STGR event. The plant measured data for pressure and the B-loop temperature were used and no credit was taken for warm prestressing. The results of this analysis showed that, for a critical flaw size of 0.91 inches, crack extension and arrest would still occur for a vessel RT_{NDT} (nil ductility transition reference temperature) value of 378°F. Based on the conservatively estimated RT_{NDT} value of 225°F for the Ginna vessel, there was considerable margin available at the time of the event. Downcomer fluid temperatures of 100°F less than the B loop measured fluid temperatures would not result in pressurized thermal shock.

Petitioner's Assertion and Request

B. Safety valve. The Sierra Club considers staff response regarding the safety significance of the steam generator safety valve malfunction and the lack of any proposed corrective action to be an unacceptable response to the Club petition #1lb. We wish to bring this concern to the Commission's attention.

The Task Force, appointed by the Commission, determined that the safety valve opened and closed five times. Staff in NUREG-0916 notes the Task Force findings regarding the malfunction of the valve in the following passage:

(3) "Fracture-Mechanics Analysis for Several PWR Recorded OCA Transients," R. D. Cheverton, D. G. Ball, S. K. Iskander, ORNL, July 20, 1982, Revised 7/27/82.

> "NUREG-0909 also notes that the valve opened and closed at generally decreasing pressures and discussed a possible reason for the decreasing closing pressures; the possibility of some steam leakage after closing the first time, and water leakage estimated at 100 gpm after the last closing. The NUREG attributed the water leakage to the likelihood of failure to fully reseat after the last closing until 50 minutes later when the valve apparently stopped leaking." (NUREG 0916 at 6-11)

Despite this release of approximately 500 gallons of cooling water contaminated via the tube rupture and released directly to the environment, the staff concludes "that the valve behavior was entirely within its design basis," (Ibid at 6-12) and that "The performance of the steam generator safety valve that opened was satisfactory." (Ibid. at 6-14). The Sierra Club is shocked by staff's conclusions. When the safety valve leaks or sticks open, there is no way operators can close the valve manually. Nor can a block valve be closed. During a SGTR accident, the safety valve is a direct path for loss of radioactive steam or water to the environment. The potential for exceeding Part 100 release limits during a design basis SGTR accident is discussed in the next section. Given this scenario, staff's conclusion that the safety valve is acceptable does not serve to increase citizen confidence in the nuclear industry's ability to protect public health and safety. We are not reassured by staff's decision to give the licensee 6 months in which to review its procedures for a tube rupture with failed SG safety or relief valve. (Ibid. at 4.1.12)

If the safety valve malfunctioned while still meeting the design basis specifications, then the specifications are clearly inadequate. The Ginna reactor should not be allowed to operate without an improved safety valve.

C. Iodine release. Staff recognizes, as a result of the Ginna accident, that "the potential exists for doses [of iodine to be released] exceeding Part 100 Guidelines for a design-basis SGTR accident." (Ibid. at 8-1) As recently as June 25, 1981, staff's analysis of such an accident contained in "Systematic Evaluate [sic] Program Evaluation of a Steam Generator Tube Rupture Accident at Ginna" had not considered the possibility of substantial amounts of water and steam being released through the safety valve. The inability of staff to model possible accident parameters accurately in advance of an accident lays open to question the basis on which regulations are promulgated.

While we commend staff's caution in reducing the spiking and equilibrium concentration limits for iodine in the primary coolant, we note that staff is willing to remove these stricter standards if licensee can demonstrate that steam generator flooding will not occur. (Ibid. at 8.1) Yet the steam generator did flood with water when it was not expected to do so. At the very least there should be a "lesson learned" from the Ginna accident that such flooding should be part of a design basis SGTR accident.

Response:

Accurate analysis of a steam generator tube rupture is complex because it involves thermohydraulic transients in the primary and secondary coolant systems that affect each other, operator actions necessary to mitigate the consequences of the accident, and a variety of ways in which the accident can evolve. It is only necessary that such accidents be analyzed conservatively. Because of this complexity, the most accurate prediction that the staff can make "in advance" is that no two steam generator tube rupture (STGR) accidents are likely to be the same. The existing SGTR accident experience supports this.

For the purposes of analyzing a design basis SGTR (like the June 25, 1981 staff analysis for Ginna), the staff makes simplifying but conservative assumptions as to the course of the accident and the pathways for the release of radioactivity. The assumptions are based on engineering judgement as to what the worst credible accident would be. The radiological consequences calculated using these assumptions, and the methodology described in Standard Review Plan (SRP) 15.6.3, "Radiological Consequences of Steam Generator Tube Rupture Accidents," are judged by the staff to be conservative, in the sense that the best estimate of doses (and doses from actual accidents) would be far less. This is because the values assumed for many accident parameters, to which the calculated dose is directly proportional, are far higher than the most probable values. Examples are iodine concentrations in the reactor coolant and the atmospheric dispersion coefficient. However, there may be some aspects of the longer-term evolution of the thermohydraulic transients that have received little attention by the staff. In particular, the type of and timing of operator actions to mitigate the accident after half an hour (or an hour) have not been evaluated in depth by the staff. These operator actions can determine, among other things, whether or not the steam generators will overfill. Also the staff currently assumes that the atmospheric dump valve and safety valves of the affected steam generator work as designed. However, during the Ginna event, the safety valve opened at successively lower pressures, finally failing to fully reseat. Although this affected the course of the incident by prolonging the leakage, the safety valve performed its design function of providing over-pressure protection of the steam generator.

The overall effect of these operator actions and equipment malfunctions on the predicted accident behavior is still under study. However, after the Ginna accident the staff re-evaluated offsite doses for a future postulated SGTR, assuming essentially no mitigative actions by the operator to stop primary-to-secondary leakage (NUREG-0916 Section 8). The results of the evaluation showed that with the new iodine concentration limits required by the staff and discussed in NUREG-0916, doses would be less than 10 CFR Part 100 guidelines, even if there was extended primary-to-secondary leakage and long-term overfill of the steam generator. It is reasonable to assume that some action to mitigate leakage would be given high priority following an actual accident, particularly if sampling showed that the reactor coolant iodine levels were as high as those assumed by the staff when calculating doses. In every past SGTR accident, the operators have taken action to reduce pressure and control leakage, even though these actions resulted in leakage beyond the times typically assumed for a design basis SGTR. The staff's assumption of no operator action is very conservative, yet it bounds the worst credible consequences, and is necessary to assure the public health and safety until the staff and licensee complete a more in-depth analysis. The staff required that the licensee re-analyze the SGTR for Ginna, giving particular attention to long-term mitigation of the accident, operator actions, and equipment malfunctions not previously examined.

The staff will carefully evaluate the re-analysis, and will not grant an increase in coolant iodine concentration technical specification limits unless the new limits and predicted plant behavior result in offsite doses less than 10 CFR Part 100 guidelines.

Petitioner's Assertion and Request

C. ... We note that staff again avoids dealing with the fact that the safety valve is not designed to handle water, or to be cycled open and closed. Staff suggests that the steam generator PORV is better suited for cycling and so "may be better to use." (Ibid. at 8-3) However, staff concedes earlier in its discussion that the relief valve is also subject to malfunction. They state:

> "Two-phase flow through the relief or safety valves may contribute to valve degradation and possible failures to reseat. This can contribute to the radiological consequences by providing a prolonged pathway to the environment." (Ibid. at 8.1, emphasis added.)

Thus, simply changing the emergency operator guidelines to ensure that the block valve is not closed incorrectly will not remedy the problem.

Response:

The ability of the safety or relief valves to pass water or a two phase mixture without degrading their performance is important in the mitigation of a SGTR if the steam generator water level becomes excessive. During the Ginna event, continued safety injection led to overfilling of the steam generator, safety valve lifting, and subsequent maloperation. As NUREG-0916 states, degraded relief or safety valve performance may contribute to offsite consequences by continuing releases.

The damaged steam generator safety valve opened five times (NUREG-0916, pg. 6-10) at successively lower pressures. The licensee asserted that the valve performance was not unexpected, and that variation in lifting pressure and blowdown may be expected due to heating of the valve internals and spring relaxation with repeated openings. However, the failing to fully reseat and the valve degradation that the licensee reported may have been due to the valve being subjected to two-phase and liquid releases. It is this latter performance, in particular, that has the most direct impact on the SGTR accident.

A number of recommendations for both the industry and the staff are in the final stages of agency review and value/impact analysis. One of the tasks proposed for the agency is to assess the probability and consequences of steam generator overfill as a result of operator errors or equipment malfunctions during a SGTR accident. As a part of this task, the staff will assess the need for qualifying the safety and relief valves for water and two-phase releases. This assessment will factor in the results of the overfill analysis, the offsite consequences as a result of a various operator errors, and the recent pressurizer PORV and safety valve testing program conducted by EPRI.

Petitioner's Assertion and Request

C. ...Staff has approved other changes which relate to termination of the safety injection. We are concerned that these changes may have ramifications for core cooling. We are particularly concerned about the following note to be added after STEP 3.15.3:

> "Termination of SI with suspected voids in the upper RV head is allowed when natural circulation is verified." (Ibid. at 8.1)

The Ginna accident has demonstrated how difficult it can be to verify natural circulation. We find no analysis of the consequences of terminating SI with a vessel void, if operators make an error in verifying natural circulation. Nor do we find any analysis of possible adverse consequences of adding STEP 3.20.3 which requires that operators "Block SI before the faulted S/G drops below 550 psig."

Response:

The Ginna event did not demonstrate any difficulty in verifying natural circulation. Following manual trip of the reactor coolant pumps, the operators, as instructed by plant procedure O-8, Revision 2, "Natural Circulation in the RCS," confirmed that natural circulation had been established by observing various plant parameters, as:

1. Loop "A" T (differential temperature) less than full power T.
2. Core exit thermocouples subcooled and constant or decreasing in temperature.
3. A-steam generator level in the narrow range, as soon as the level recovered from the reactor trip.
4. Auxiliary feed flow to A-steam generator.

It is highly unlikely that, given the above plant parameters, the operators can make an error in verifying natural circulation. Nevertheless, in the unlikely event that natural circulation is not established, termination of safety injection (SI) with a vessel void would result in a gradual repressurization of the reactor coolant system. The repressurization of the reactor coolant system and reversal in direction of the four plant parameters listed above is an indication to the operators that natural circulation has not been achieved, and the procedures direct the operators to alternative methods for depressurizing and cooling the primary system.

In step 3.20.3 of procedure E-1.4, the operators are instructed to "block SI before the faulted S/G drops below 550 psid," in order to preclude inadvertent actuation of SI by the faulted S/G low pressure SI actuation setpoint. In the event, however, of an actual need for SI, following the block of the faulted S/G actuation variables, the redundant primary system variables or the intact S/G pressure variables will independently initiate SI.

<u>Petitioner's Assertion and Request</u>

C. ...Staff admits that there has been "incomplete evaluation of the effects of changes to operator guidelines," (<u>Ibid.</u>) which is one reason the iodine limits are being lowered. The Sierra Club urges the Commission to reconsider the wisdom of allowing Ginna to restart when operating guidelines have been changed without complete evaluation of the safety repurcussions [sic] of these changes.

Response:

The staff's evaluation of the procedural improvements made by the licensee in response to the SGTR are contained in Section 4 of NUREG-0916. Based on the licensee's response to the event and the subsequent program for further improvements, the staff concluded that adequate protection is provided for steam generator tube rupture events. The licensee committed, at that time, to study further the areas of pump trip and restart, cooldown of a faulted steam generator, coping with a reactor vessel steam bubble, and additional natural circulation cooldown guidance. The staff will review these studies when they are submitted and any further modifications to Ginna's procedures resulting from these studies will be included in the review.

As stated previously, after the Ginna accident the staff re-evaluated offsite doses for a future postulated SGTR, assuming essentially no mitigative actions by the operator to stop primary-to-secondary leakage. The results of the evaluation showed that with the new iodine concentration limits recommended by the staff, doses would be less than 10 CFR Part 100 guidelines, even if there was extended primary-to-secondary leakage and long-term overfill of the steam generator. It is reasonable to assume that some action to mitigate leakage would be given high priority following an actual accident, particularly if sampling showed that the reactor coolant iodine levels were as high as those assumed by the staff when calculating doses. The staff's assumption of no operator action is very conservative, yet it bounds the worst credible consequences, and will assure the public health and safety until the staff and licensee complete a more in-depth analysis.

Petitioner's Assertion and Request

D. Steam Generator Tubes. In response to concerns raised in Sierra Club's petition at #2a, b, c and #3 regarding in-service inspection standards and specifications for tube rejection, staff simply renumerates the current standards and RG&E procedures. There is no recognition by staff that the inability to anticipate the January 25th tube burst, despite recurrent problems in wedge area #4 and eddy current indication in April, 1981, for the tube that later burst, should be a warning that the standards are not adequate. The Sierra Club is concerned that staff has avoided dealing with the implications of the tube burst and urges the Commission to review the adequacy of these standards.

Response:

The adequacy of the eddy current test procedures, data evaluation, and calibration standards were reviewed by the NRC staff and by an expert consultant to the staff who was present at the Ginna site. The results of this review and our conclusions are described in detail in Section 5.2.4.1, 5.3.1.2, 5.4.3 of the staff's SER (NUREG-0916).

The immediate cause of the tube rupture occurrence was excessive tube wall penetration by a smooth fretting type wear flaw which lead to a pressure burst of the tube. Such a smooth or gradually tapered flaw may produce little or no signal on the differential channels depending on the degree of smoothness or taper. This type of flaw will produce a detectable signal on the absolute data channels. However, the staff believes that special calibration standards with simulated wear defects should be employed in addition to the standards required by the ASME Code to ensure a conservative interpretation of signals produced by such defects and is including these standards in its generic review of the Ginna event.

Calibration standards with simulated wear flaws had not been used during the previous inspection in April 1981. The tube which later ruptured in January 1982 had not exhibited a differential signal in April 1981, but did exhibit an absolute signal which was interpretable as less than a 20% through-wall penetration using ASME Code calibration standards. Given the present knowledge that the tube was degraded by a smooth fretting type wear flaw, the less than 20% interpretation of the April 1981 signal is likely to be non-conservative. This signal is interpretable as a slightly greater than 40% through-wall indication using calibration standards with a simulated wear flaw. Thus, we expect that this tube would have been plugged in April 1981 had this standard been used to evaluate the signal on the absolute channel.

The eddy current inspections conducted subsequent to the rupture occurrence employed both differential and absolute mode inspection. Wear calibration standards were also employed during this inspection. We believe these inspections were adequate to detect any tubes with the type of flaw which caused the tube rupture.

Regarding the 40% plugging limit, the limit has been developed to assure that there is sufficient remaining wall thickness to preclude rupture over the full range of normal and postulated accident conditions. This limit makes allowance for approximately 10% additional throughwall penetration prior to performing the next inspection of the tube. This allowance is generally adequate based upon operating experience. However, due to the presence of loose parts, the degradation rate for the tube which ruptured was apparently much higher than what is allowed for in the plugging limit. Thus, it is necessary to eliminate the conditions for continuing the degradation mechanisms which led to the rupture, in addition to performing eddy current inspections and plugging ..ose tubes that exceed the plugging limit. This was the objective of the repair program conducted at Ginna following the rupture occurrence. The repair program (discussed in Section 5.5 of the staff's SER) included the removal of all foreign objects and loose parts are the removal of previously plugged tubes which could potentially cause damage to adjacent tubes. Thus, we do not expect further progression of the impact and wear damage from foreign objects which had been occurring for several years up to January 25, 1982.

Petitioner's Assertion and Request

E. PORV. The Sierra Club raised the concern that the PORV is not required to be safety grade in its petition at #7 and asked for staff review in light of the Ginna accident and the failure of the PORV. Staff has responded that a generic study is underway. (Denton response of May 22, page 5) The fact that a specific cause has been determined for the Ginna PORV failure in no way obviates the importances of making the PORV safety grade. How many accidents involving a malfunction of the PORV need to take place before the staff determines that these valves need to be upgraded? This question is ripe for Commission consideration.

Response:

It is uncertain whether upgrading the PORV to safety grade will provide the desired improvement in the ability of a PORV to reclose following an automatic or manual opening. Operability tests conducted by EPRI on PORVs, similar to those conducted for safety valves which are safety grade, have demonstrated acceptable performance. However, some failures to reclose have continued to occur in power plants.

Although PORV failures are undesirable from an operational standpoint, it is not yet clear whether such failures pose an unacceptable risk to public health and safety. For example, if PORV failures are not considered to increase the probability of core melt, then upgrading may not be warranted. The staff study acknowledged in the May 22 Director's Decision is nearing completion and the staff's recommendations will be presented when the study is completed.

III.

Ms. Caplan urges, "Where generic investigations are not already underway, we hope that the Commission will institute such proceedings so that the "lessons learned" from the Ginna accident will not be lost."

The Commission staff has initiated a study of the matters affecting steam generator tube degradation and steam generator tube rupture (SGTR) events which may have generic applications. The scope of the information being considered for these studies includes the Ginna STGR as well as three previous domestic SGTR's, the results of ongoing staff studies regarding tube degradation, and recent steam generator operating experiences, including foreign experiences, where available. Results of this study may fall into one of three areas: (1) they could be applicable to already ongoing staff generic efforts and the lessons learned from the study are therefore planned to be factored into those ongoing studies, (2) the results could define areas which require further evaluation by the staff prior to determining the actions needed to respond to the subject, and (3) the results might be identified as candidates for generic application to all pressurized water reactors and are therefore being subjected to value/impact analyses and further review by the staff to determine which candidates will be applied as generic requirements. The process for this latter category is currently underway and is expected to be completed in late 1982.

For the reasons and under the conditions described in the staff's restart SER (NUREG-0916), the R. E. Ginna plant can be operated without undue risk to public health and safety. Although additional analyses and studies of such issues as pressurized thermal shock, steam generator degradation and tube rupture transients are underway, Ms. Caplan's letter provides no new information that would lead the staff to alter its conclusions in NUREG-0916 or that would require suspension of plant operation pending the completion of ongoing and planned studies. Therefore, I have determined that no adequate basis exists for ordering the suspension of the operating license for the R. E. Ginna Nuclear Power Plant. Consequently, Ms. Caplan's request is denied.

A copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c). As provided in this regulation, the decision will become the final action of the Commission twenty-five (25) days after issuance, unless the Commission, on its own motion, institutes review of the decision within that time.

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 8th day of October 1982.

Attachment:
Figure 1

TEMPERATURE DISTRIBUTION IN A VESSEL WALL

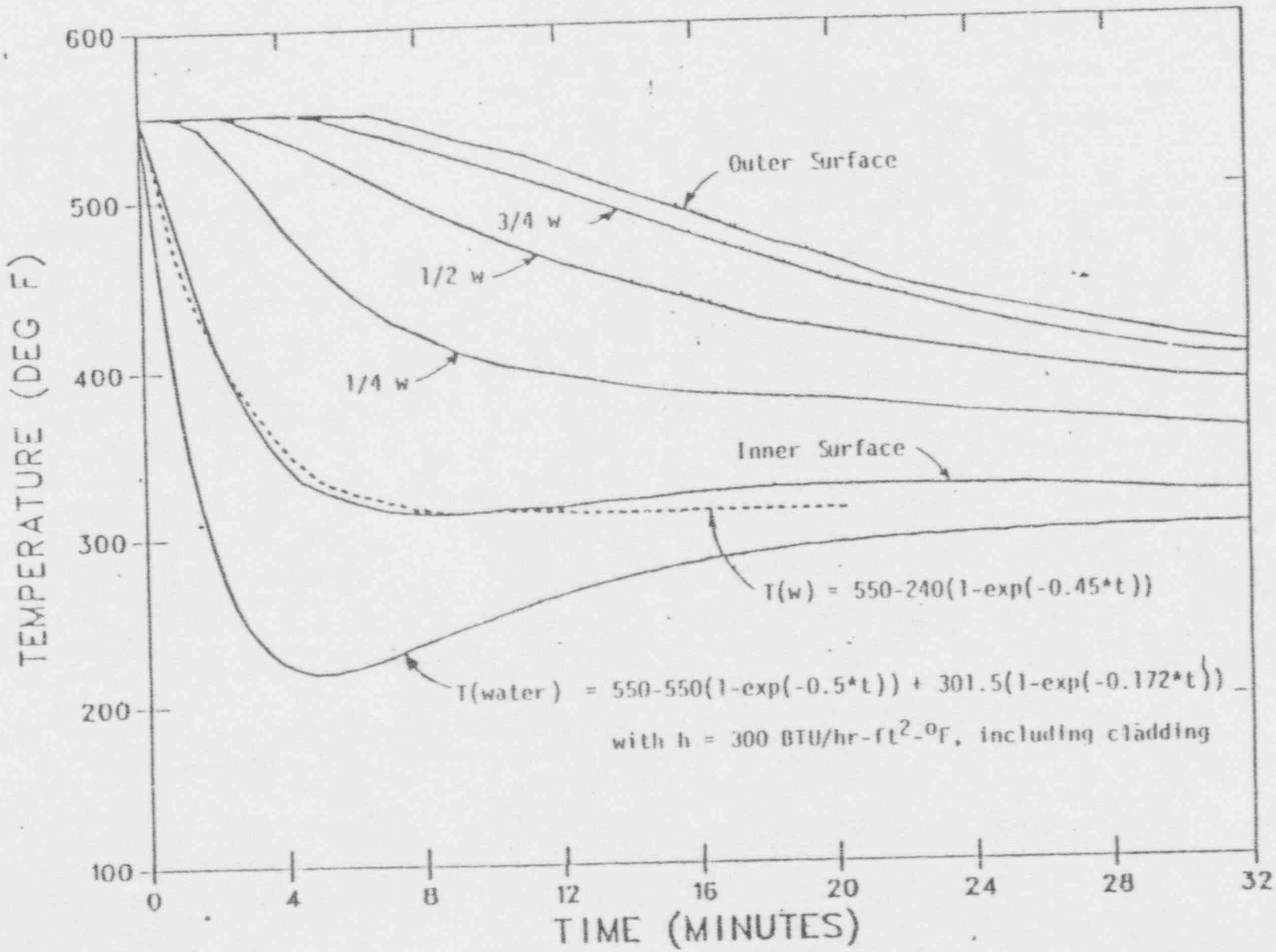

FIGURE 1

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

ISSUANCE OF DIRECTOR'S DECISION

UNDER 10 CFR SECTION 2.206

On March 11, 1982, (47 FR 14988, April 7, 1982), Ms. Ruth Caplan, Chair, Sierra Club, filed a show cause petition with the Nuclear Regulatory Commission's (NRC) Office of Nuclear Reactor Regulation (the staff) requesting that the operating license for the R. E. Ginna Nuclear Power Plant, located in Wayne County, New York, be suspended or, in the alternative, permission to restart the reactor be withheld, until critical safety issues were reviewed relating to the January 25, 1982, steam generator tube rupture event. The petition was considered under 10 CFR Section 2.206.

On May 22, 1982, the Director of Nuclear Reactor Regulation denied the portion of Ms. Caplan's request relating to suspension of operation. However, the Director granted the petitioner's request that the staff review include and consider specific areas detailed in the petition prior to restart of the Ginna plant. The documentation of this review is contained in NUREG-0916 [See Director's Decision DD-82-03, 15 NRC _____ (May 22, 1982)].

By letter dated June 10, 1982 Ms. Caplan requested that the Commission exercise its authority under 10 CFR 2.206(c) to review the issues raised in the petition dated March 11, 1982. The Commission referred Ms. Caplan's June 10 request to the staff for consideration under the provisions of 10 CFR Section 2.206 of the Commission's regulations.

Upon review of information pertaining to the concerns at the Ginna plant and the information provided by Ms. Caplan, the Director of Nuclear Reactor Regulation has determined that issuance of an order to show cause why the operating license for the Ginna plant should not be suspended is not warranted. Accordingly, Ms. Caplan's June 10, 1982 request has been denied.

The reasons for the denial are explained in the "Director's Decision" under 10 CFR 2.206 (DD-82-11), as supported by DD-82-03 dated May 22, 1982 and the safety evaluation contained in NUREG-0916, which are available for public inspection in the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Local Public Document Room at the Rochester Public Library, 115 South Avenue, Rochester, New York 14604.

A copy of the decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c). As provided in this regulation, the decision will become the final action of the Commission twenty-five (25) days after issuance unless the Commission, on its own motion, institutes review of this decision within that time.

Dated at Bethesda, Maryland, this 8th day of October 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

October 25, 1982 SECY-82-437

ADJUDICATORY ISSUE
(Affirmation)

For: The Commission

From: Martin G. Malsch, Deputy General Counsel

Subject: PENDING COMMISSION PROCEEDING CONCERNING RENEWAL OF BYPRODUCT MATERIALS LICENSE OF SELF-POWERED LIGHTING, INC.

Discussion: On April 2, 1982, the Commission instituted a proceeding to consider whether the staff had decided correctly that the byproduct materials license of Self-Powered Lighting, Inc., (SPL) should not be renewed. Under its license, SPL, which is located in the Agreement State of New York, is authorized to distribute self-luminous gunsights containing tritium under 10 CFR §§ 30.19 and 32.22.

As is explained in the attached proposed order, the staff originally had indicated that because SPL sought renewal with a condition limiting distribution to certain users, the State of New York was the proper licensing authority. In the course of the proceeding, however, the staff decided to renew the existing SPL license and

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Paul Bollwerk, GC
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include the condition limiting distribution. The staff also asked that the Commission's proceeding be dismissed as moot.

We believe that

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Recommendation:

Martin G. Malsch
Deputy General Counsel

Attachment:
Proposed Order

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Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Wednesday, November 10, 1982.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Wednesday, November 3, 1982, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of November 15, 1982. Please refer to the appropriate Weekly Commission Scheduled, when published, for a specific date and time.

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