MAR 7 1974

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Edward J. Nelson, President
89 East Avenue
Rochester, New York 14604

Change No. 11
License No. DPR-18

Gentlemen:

By letter dated May 23, 1973, and supplement dated November 12, 1973, you submitted an application for several changes to the Technical Specifications appended to License No. DPR-18 for the Robert Emmett Ginna Unit No. 1. The proposed changes result from information gained in the testing and analysis of the specimens irradiated in the first reactor vessel surveillance capsule and include heatup and cooldown curves for normal operation, minimum conditions for criticality, a revised inspection schedule for inservice radiation capsules, and minimum temperatures for performing inservice leak and hydrostatic tests.

We have completed our review of the proposed changes with minor modifications that have been discussed with members of your staff and have designated our action as Change No. 11. A copy of our Safety Evaluation supporting this action is attached.

We have also included in our action, as discussed with members of your staff, certain changes to the Plant Reporting Requirements in Section 6.6 of the Technical Specifications to meet regulatory requirements. These specific changes are required to bring the Technical Specifications in conformance with the Regulatory Guide 1.16, "Reporting of Operating Information" (Revision 1 issued October 1973).

We conclude that Change No. 11 does not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. Pursuant to 10 CFR Part 50, Section 50.59, the Technical Specifications appended to License No. DPR-18 are changed as shown in Attachment A.

Sincerely,

Original signed by
Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

CP

Enclosures: See next page

OFFICE

Enclosures:
1. Attachment A - Change No. 11 to the Technical Specifications
2. Safety Evaluation

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

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

Rochester Committee for
Scientific Information
Robert E. Lee, P.H.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

Monroe County Conservation
Thomas R. Burns
Harris, Carroll and Creary
226 Power Building
Rochester, New York 14614

Dr. William Seymour
Staff Coordinator
New York State Atomic Energy Council
New York State Department of Commerce
112 State Street
Albany, New York 12207

~~Mr. Hans L. Hamester (2 copies)~~
~~ATTN: Joan Sause~~
~~Office of Radiation Programs~~
~~Environmental Protection Agency~~
~~Room 647A East Tower, Waterside Mall~~
~~401 M Street, S. W.~~
~~Washington, D. C. 20460~~

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

Lyons Public Library
67 Canal Street
Lyons, New York 14489

DISTRIBUTION
Docket File(50-244)
AEC PDR
RP Reading File
Branch Reading File
JRBuchanan, ORNL
TBAbernathy, DTIE
ACRS (16)
RO (3)
DJSkovholt, L:OR
RLTedesco, L:CS
RJSchemel, L:ORB#1
RHVollmer, L:QAB
PCollins, L:OLB
TJCarter, L:OR
RRRenfrow, OGC
SVarga, L:RP
TVWambach, L:ORB#1
NDube, L:OPS
MJinks, DRA (4)
BScharf, DRA (15)
SKari, L:RP
SATeets, L:ORB#1

OFFICE	L:ORB#1 DEElliott:dc	L:ORB#1	L:ORB#1	L:OR		
SURNAME	TVWambach	SATeets	RJSchemel	DJSkovholt		
DATE	3/4/74	3/5/74	3/6/74	3/7/74		

ATTACHMENT A

CHANGE NO. 11 TO THE TECHNICAL SPECIFICATIONS

LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

OFFICE➤						
SURNAME➤						
DATE➤						

1. Section 1.9 - Replace subparagraphs c. and f. with the following:

"c. An uncontrolled or unplanned release of radioactive material from the site boundary."

"f. Uncontrolled or unanticipated changes in reactivity equal to or greater than 1%$\Delta k/k$."

2. Section 3.1 - Pages 3.1-5 through 3.1-19 are to be replaced with the revised pages attached.

3. Section 4.2 - Page 4.2-3 is to be replaced with the revised page attached.

4. Section 4.3 - Page 4.3-1 is to be replaced with the revised page attached and a new page numbered 4.3-3 added.

5. Section 6.6 - Pages 6.6-1, 6.6-6, and 6.6-7 are to be replaced with the revised pages attached and a new page numbered 6.6-6a added.

OFFICE➤						
SURNAME➤						
DATE➤						

3.1.2 Heatup and Cooldown Limit Curves for Normal Operation

3.1.2.1 The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2 for the first 7.0 effective full power years.

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr, respectively. Limit lines for cooldown rates between those presented may be obtained by interpolation.

b. Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. The limit lines shown in Figures 3.1-1 and 3.1-2 shall be recalculated periodically using methods discussed in the Basis Section.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator vessel is below 70°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis: Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965

Section III of the ASME Boiler and Pressure Vessel Code, Reference (1), and ASTM E185, Reference (2), and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code, Reference (3), and the calculation methods described in Reference (4).

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 7 years of service life. The 7-year service life period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table 3.1-1. The fast

---

(1) ASME Boiler and Pressure Vessel Code Section III (Summer 1965)

(2) ASTM E185 Surveillance Tests on Structural Materials in Nuclear Reactors

(3) ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda (note Code Case 1514)

(4) W. S. Hazelton, S. L. Anderson, and S. E. Yanichko, WCAP-7924, "Basis for Heatup and Cooldown Limit Curves"

neutron (E>1 Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1-4. Using the applicable fluence at the end of the 7-year period and the copper content of the material in question, the ΔRT_{NDT} is obtained from Figure 3.1-3. The ΔRT_{NDT} is more conservative than the value obtained from the first capsule of radiation surveillance program.

Values of ΔRT_{NDT} determined in this manner will be used until more results from the material surveillance program, when evaluated according to ASTM E185, are available. The next capsule will be removed early in the service life of the reactor vessel, note Radiation Surveillance Section in FSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is greater than the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Heatup and Cooldown Limit Curves

Allowable pressure temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Codes and discussed in detail in Reference (4).

The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown in the K_{IR} curve for the metal temperature at that

OFFICE ➤
SURNAME ➤
DATE ➤
Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

time. Furthermore, the approach applies explicit safety factors of 2.0 and 1.25* on stress intensity factors induced by pressure and thermal gradients, respectively. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\ K_{Im} + 1.25\ K_{It} \leq K_{IR}$$

where: K_{Im} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that during heatup the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

* The 1.25 safety factor on K_{IT} represents additional conservatism above Code requirements.

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep OD flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under construction. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

Change No. 11
Date: MAR 7 1974

OFFICE➤						
SURNAME➤						
DATE➤						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 · 520-294

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses ath the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature; whereas, the limiting pressure is calculated using the material temperature at the tip of the

OFFICE➤						
SURNAME➤						
DATE➤						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition is, of course, not true for the steady state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in Reference (4).

Temperature requirement for the steam generator corresponds with the measured NDT for the shell of the steam generator.

A temperature difference of 320°F between the pressurizer and reactor coolant system maintains thermal stresses within the pressurizer spray nozzle below design limits.

KEUFFEL & ESSER CO.

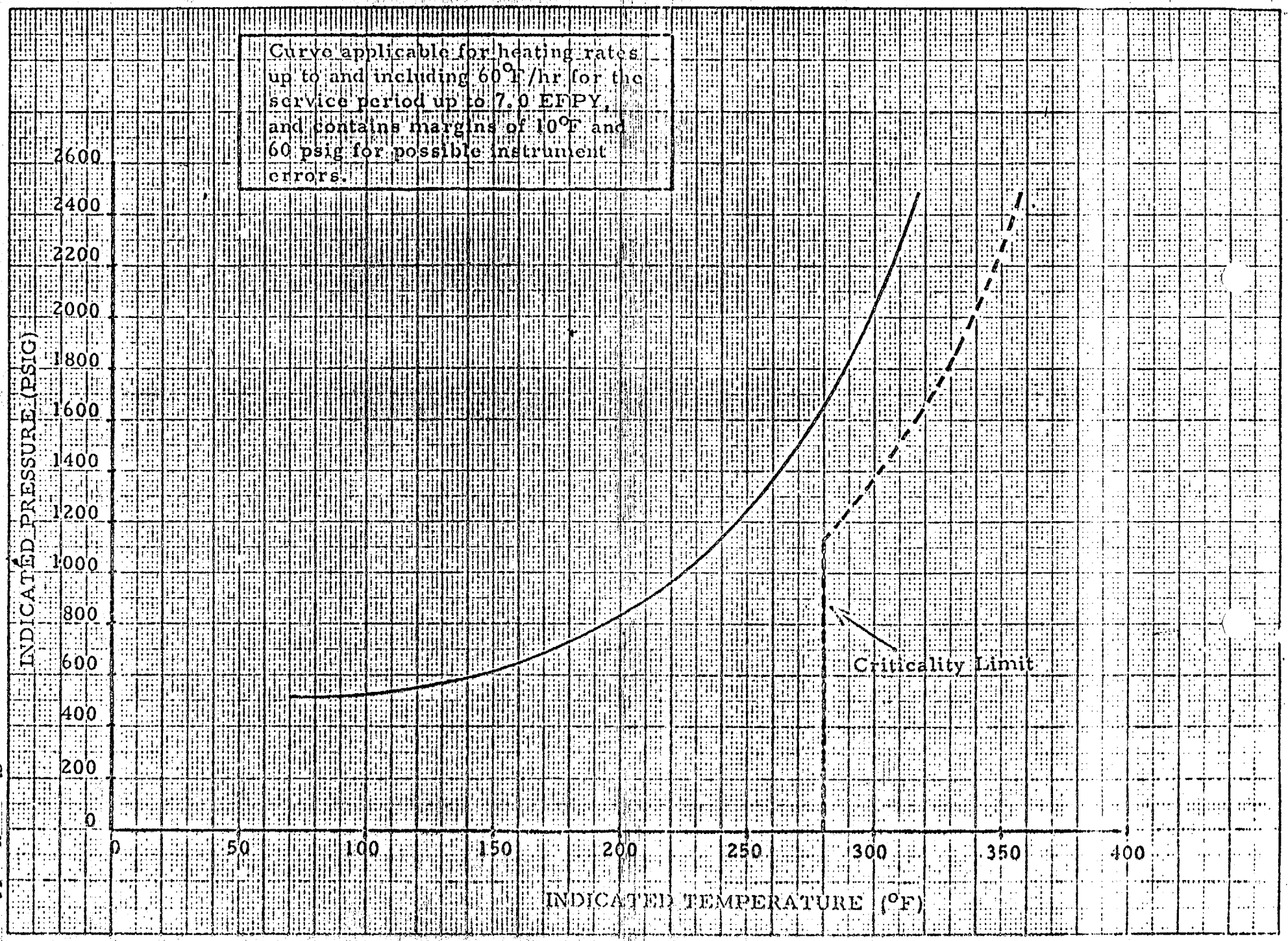

Figure 3.1-1 RG&E Heatup Limitations Applicable to 7.0 Effective Full Power Years

10 X 25 CM. MADE IN U.S.A.
KEUFFEL & ESSER CO.

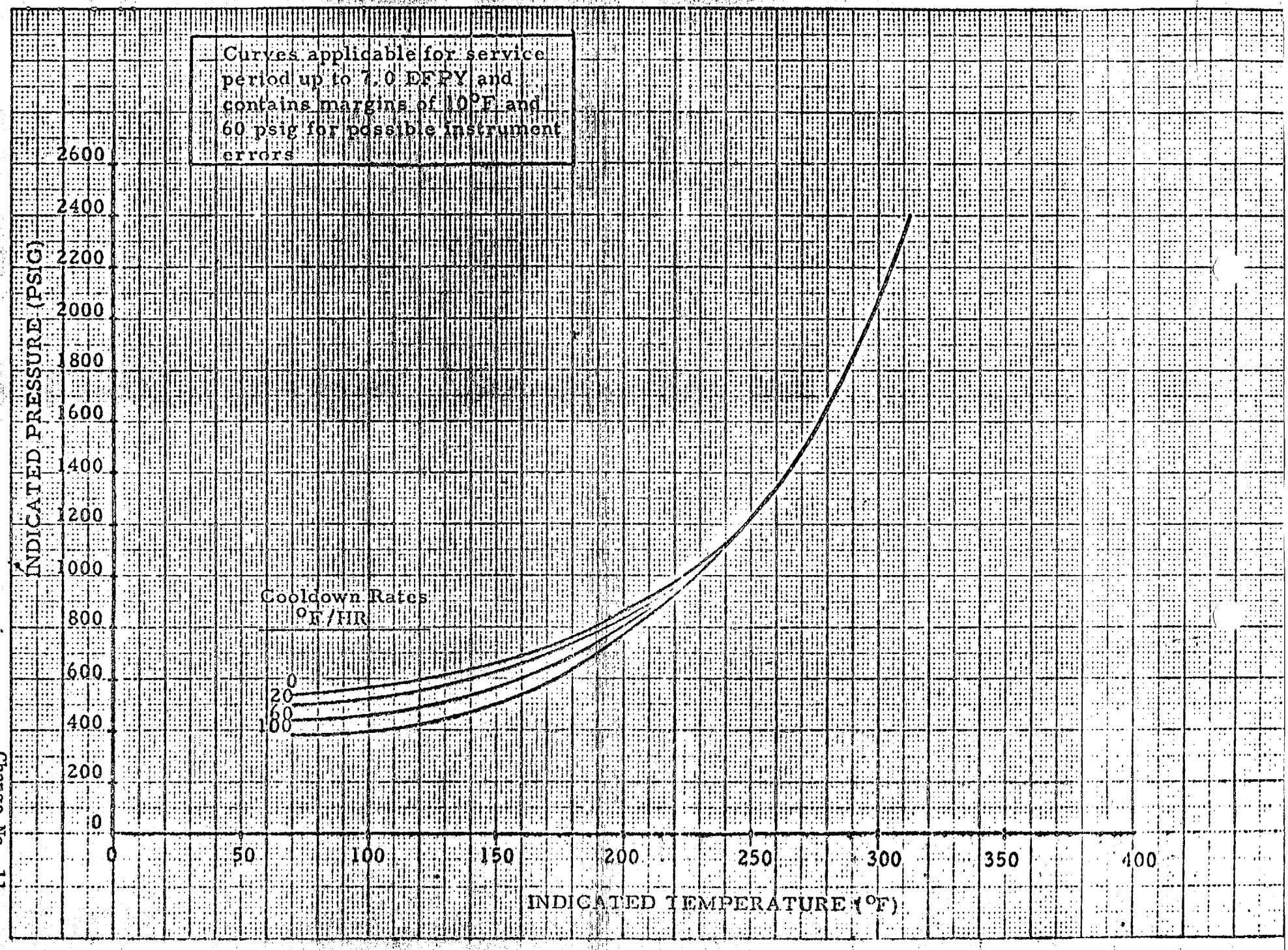

Figure 3.1-2 RG&E Cooldown Limitations Applicable to 7.0 EFPY

Change No. 11
Date: MAR 7 1974

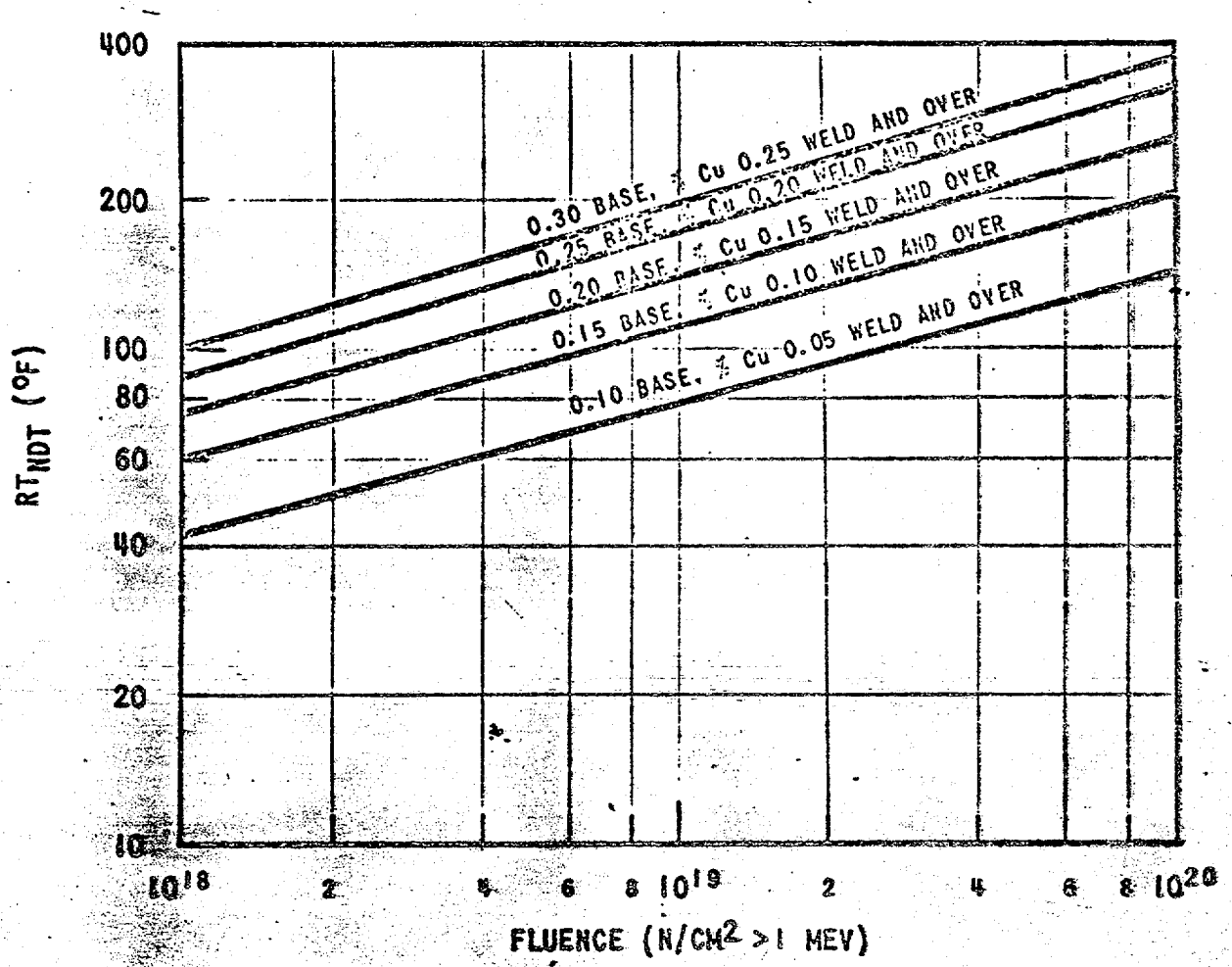

Figure 3.1-3 Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel - Steels Exposed to 550°F Temperature

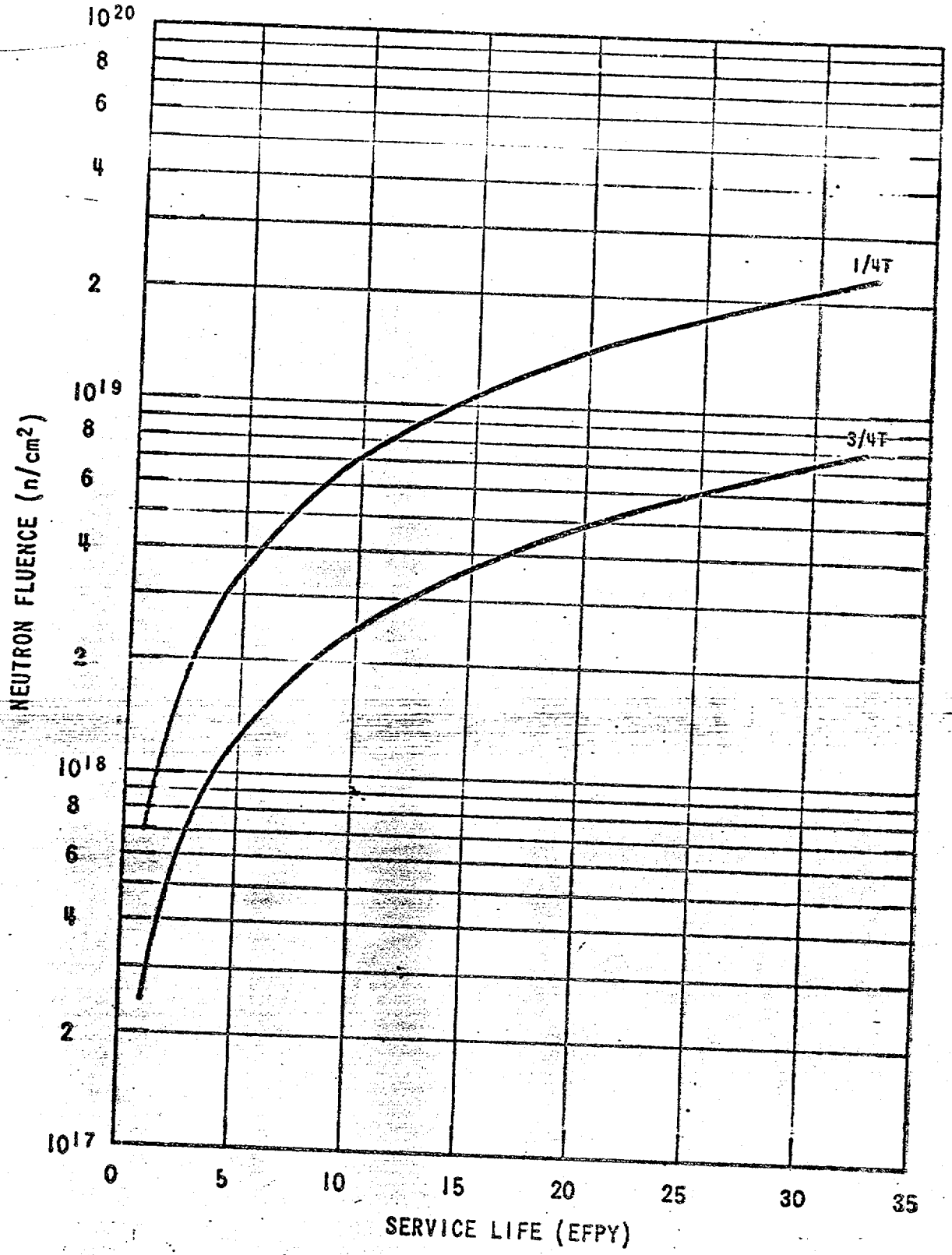

Figure 3.1-4 Fast Neutron Fluence (E > 1 Mev) as a Function of Full Power Service Life

Change No. 11
Date: MAR 7 1974

TABLE 3.1-1

RGE REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

COMPONENT	MATL TYPE	Cu (%)	P (%)	NDTT (°F)	50 FT-LB/35 MILS LATERAL EXPANSION TEMP (°F) LONG	TRANS	RT_{NDT} (°F)	UPPER SHELF (FT-LB) LONG	TRANS
CL.HD.DOME	A302,B	NA	0.10	-20	52	125**	65	NA	NA
CL.HD.FLG	A508,2	NA	0.007	-75	-50	-35**	-75	140*	91***
VES.SH.FLG	A508,2	NA	0.007	-82	-30	8**	-52	115*	75***
INJEC.NOZ	A508,2	NA	0.013	60*	65	95**	60	66	43***
INLET NOZ	A508,2	0.09	0.012	60*	-20	35**	60	107	70***
INLET NOZ	A508,2	0.09	0.010	60*	-22	15**	60	115	75***
OUTLET NOZ	A508,2	0.09	0.011	60*	0	45**	60	109	71***
OUTLET NOZ	A508,2	0.09	0.012	60*	22	60*	60	88	57***
UPPER SHL	A508,2	NA	0.010	40	12	20**	40	173	112***
INTER SHL	A508,2	0.07	0.010	20	40	105**	45	120	78***
LOWER SHL	A508,2	0.05	0.010	40	10	30**	40	164	107***
TRANS.RING	A508,2	NA	0.008	10	-30	-13**	10	182	118***
BOT.HD.DOM	A302,B	NA	0.010	-50	35	85**	25	NA	NA
WELD	Weld	0.23	0.012	0*	NA	45	0	NA	74
HAZ	HAZ	NA	NA	NA	55	NA	NA	77	NA

*No data, conservative estimate

**Estimated from longitudinal data, temperature at which minimum 77 ft-lb/55 mils lateral expansion obtained

***Longitudinal value reduced 35%

Change No. 11
Date: MAR 7 19

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at a temperature below 500°F if the moderator temperature coefficient is positive.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit line shown on Figure 3.1-1 of these specifications.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.[1] [2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the initial fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the initial fuel cycle and during subsequent reload fuel cycles, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[1] [2] At the beginning of life

of the initial fuel cycle, the maximum temperature at which the moderator coefficient is positive, with all control rods withdrawn, is about 540°F.[2] When control rods are inserted, the temperature at which the moderator coefficient becomes negative is lower, so that at 500°F with the operational control rod program the temperature coefficient is expected to be negative. However, even with the rods-out moderator coefficient, the maximum potential reactivity insertion that could result from heating or depressurizing the coolant from 500°F is .001 $\Delta k/k$.[2] This is only about 15% of prompt critical and well below values (.004 and .005 $\Delta k/k$) evaluated to be safe for similar reactors.[3] [4]

The requirement that the reactor is not to be made critical at a temperature below 500°F when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient [5] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical above and to the left of the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of an increase of moderature of temperature or a decrease of coolant pressure.

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-8

(3) Amendment 14 to Connecticut Yankee License Application, Docket No. 50-213, March 2, 1967

(4) Southern California Edison Co. filing, Docket No. 50-206, October 19, 1967

(5) FSAR Figure 3.2.1-10

TABLE 4.2-1 (CONTINUED)

Component & Examination Area	Sample	Inspection Process (See Note 1)	Extent & Frequency (See Note 2)
b. Circumferential and Longitudinal Pipe Welds Branch Pipe Connections Welds	100%	V	k.
c. Integrally-Welded Supports	15%	V and UT	l.
d. Socket Welds and Pipe Branch Connections Welds 4 in. Diameter and Smaller	15%	V and S	g.
F. Reactor Coolant Pump			
a. Flywheel	100%	UT	i.
b. Supports and Hangers	100%	V	c.
G. Radiation Specimens	Capsule V	Tensile, Charpy, WOL	End of 1st core cycle
	Capsule R	Tensile, Charpy, WOL	End of 3rd core cycle
	Capsule S	Tensile, Charpy, WOL	10 years, at nearest refueling
	Capsule N	Tensile, Charpy, WOL	20 years, at nearest refueling
	Capsule T	Tensile, Charpy, WOL	30 years, at nearest refueling
	Capsule P	Tensile, Charpy, WOL	Standby
H. Valve Pressure Boundary			
a. Valve Body Seam Welds	1 valve	V	c.
I. Pressure Retaining Bolting	—	V and S and UT	h.

OFFICE ➤
SURNAME ➤
DATE ➤

Change No. 11
Date: MAR 7 1974

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

4.3 Primary System Testing Following Opening

Applicability

Applies to test requirements for primary system integrity.

Objective

To specify tests for primary system integrity after the system is closed following normal opening, modification, or repair.

Specification

4.3.1 Whenever the primary system is closed after it has been opened, the system shall be leak tested at not less than 2335 psig at temperatures above those listed in Table 4.3-1.

4.3.2 Whenever primary system modifications or repairs have been made which involve new strength welds on components greater than 2 in. diameter, the new welds shall receive both a surface and 100% volumetric non-destructive examination and shall meet applicable code requirements.

4.3.3 Whenever primary system modifications or repairs have been made which involve new strength welds on components 2 in. diameter or smaller, the new welds shall receive a surface examination.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

TABLE 4.3-1

TEMPERATURE CONDITIONS FOR HYDROSTATIC TESTS

Effective Full Power Years	Fast Neutron Fluence (E>1 Mev) At 1/4 t	$T_{1/4t}$ (°F)	Test Temp (°F)
5	3.5×10^{18}	136	274
6	4.2×10^{18}	144	282
7	4.9×10^{18}	150	288
8	5.6×10^{18}	156	294
9	6.4×10^{18}	162	300
10	7.1×10^{18}	167	305
15	1.05×10^{19}	186	324
20	1.41×10^{19}	203	341
32 (End-of-Life)	2.25×10^{19}	230	368

Change No. 11
Date: MAR 7 1974

OFFICE ➤
SURNAME ➤
DATE ➤

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

6.6 Plant Reporting Requirements

The following information shall be submitted to the USAEC in addition to the reports required by Title 10, Code of Federal Regulations.

6.6.1 Semiannual Operating Report

A report covering a six-month period shall be submitted in writing to the Director of the Regional Regulatory Operations Office within 60 days after January 1 and July 1 of each year and shall include the following:

a. Operations Summary

A summary of operating experience occurring during the reporting period that relates to safe operation of the facility, including a summary of:

1. changes in facility design,
2. performance characteristics (e.g., equipment and fuel performance),
3. changes in operating methods which were necessitated by 1 and 2 or which otherwise were required to improve the safety of facility operations,
4. results of surveillance tests and inspection required by these technical specifications,
5. the results of any periodic containment leak rate tests performed during the reporting period,

3. If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

i. Occupational Personnel Radiation Exposure

Tabulate the number of personnel exposures for facility operations personnel (permanent and temporary) in the following exposure increments for the reporting period:

less than 100 mrem, 100-250 mrem, 250-500 mrem, 500-750 mrem, 750-1000 mrem, 1-2 rem, 2-3 rem, 3-4 rem, 4-5 rem, 5-6 rem, and greater than 6 rem.

Tabulate the number of personnel receiving more than 500 mRem exposure in the reporting period according to duty function, i.e., routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine refueling operations, special refueling operations (describe operation), and other job related exposures. Annually tabulate the number of personnel receiving more than 3 rem and report major cause(s).

j. Primary Coolant Chemistry

A tabulation on a monthly basis of the maximum, average, and minimum values for the following primary coolant system parameters:

1. gross radioactivity in uCi/ml,
2. suspended solids in parts per million,
3. gross tritium in uCi/ml,
4. iodine-131 in uCi/ml,

OFFICE➤					Change No. 11 Date: MAR 7 1974
SURNAME➤					
DATE➤					

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-294

5. ratio of iodine-131 to iodine-133,

6. hydrogen in cc per kg,

7. lithium in parts per million,

8. boron-10 in parts per million,

9. oxygen-16 in parts per billion,

10. chloride in parts per million, and

11. pH at 25°C.

6.6.2 Non-Routine Reports

a. Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (cc to the Director of Licensing) followed by a written report within 10 days to the Director of the Regional Regulatory Operations Office in the event of the abnormal occurrences as defined in Section 1.9. The written abnormal occurrence report should follow the format presented in Regulatory Guide 1.16.

b. Unusual Events

A written report shall be forwarded within 30 days to the Director of the Regional Regulatory Operations Office in the event of:

1. Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the technical specifications.

OFFICE➤						
SURNAME➤						
DATE➤						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

2. Discovery of any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.

3. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

6.6.3 Special Reports

Special reports shall be submitted in writing within 90 days to the Director of Licensing, USAEC, Washington, D. C. 20545.

Special reports shall be submitted covering inspections, tests, and maintenance that are appropriate to assure safe operation of the plant. The frequency and content of these special reports are determined on an individual case basis and designated in these technical specifications. Such reports shall include:

a. Containment leak rate tests,

b. Materials radiation surveillance specimens reports.

OFFICE➤						
SURNAME➤						
DATE➤						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

SAFETY EVALUATION BY THE

DIRECTORATE OF LICENSING

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

By application dated May 23, 1973, and supplement dated November 12, 1973, Rochester Gas and Electric Corporation proposed changes to the Technical Specifications appended to License No. DPR-18 that result from information gained in the testing and analysis of the specimens irradiated in the reactor vessel surveillance program. The proposed changes include heatup and cooldown curves for normal operation based on material properties measured in the irradiated samples, minimum conditions for criticality based on the relationship between reactor coolant pressure and temperature, a revised inspection schedule for the radiation capsules remaining in the reactor vessel surveillance program, and minimum temperatures for performing inservice leak and hydrostatic tests following primary system maintenance.

We have reviewed the proposed changes and have independently determined that the irradiated properties of the shell forgings and weld metal are adequate for the safe operation of the R. E. Ginna Unit No. 1 Nuclear Power Plant. The proposed Technical Specifications changes contain acceptable pressure-temperature limits for reactor vessel heatup, cooldown, and core operation applicable for the first seven effective full power years.

The minimum conditions for criticality are presented as a function of temperature and pressure using the same graphical presentation from which heatup limitations are determined. This method for determining the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained.

A modified inservice inspection schedule for the remaining structural material radiation capsules has been presented to satisfy the requirements of 10 CFR Part 50 Appendix H.

Minimum temperatures for hydrostatic tests of the primary system following maintenance which opens the primary system have been tabulated for inclusion in the Technical Specifications. We have concluded that the proposed minimum temperatures are adequate.

OFFICE➤						
SURNAME➤						
DATE➤						

Rochester Gas and Electric Corporation has stated that they will examine the heatup and cooldown limitations and the hydrostatic test requirements and revise them if necessary following examination of the next inservice radiation capsule.

Accordingly, we conclude the proposed changes and additions to the Technical Specifications do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Elliott

Operating Reactors Branch #1
Directorate of Licensing

Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Date: MAR 7 1974

OFFICE ➤						
SURNAME ➤						
DATE ➤						