Docket No. 50-244

OCT 27 1987

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Kober:

SUBJECT: ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICNESE NO. DPR-18

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated November 10, 1983 and subsequently revised in letters dated January 21, 1986, February 13, 1987, March 9, 1987, April 14, 1987, and October 2, 1987.

This amendment revises the requirements of the Technical Specifications related to the definition of operability.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/S/

Carl Stahle, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 24 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

PDI-3 r/f	NRC PDR	Local PDR
VRooney	BBoger	MRushbrook
CStahl	OGC-Bethesda	DHagan
EJordan	JPartlow	TBarnhart (4)
Wanda Jones	EButcher	ACRS (10)
GPA/PA	ARM/LFMB	

OFC	PDI-3	PDI-3	OGC	PDI-3			
NAME	MRushbrook	CStahle		VNerses			
DATE	10/ /87	10/ /87	10/15/87	10/21/87			

OFFICIAL RECORD COPY

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

OCT 27 1987

Docket No. 50-244

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Kober:

SUBJECT: ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE NO. DPR-18

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated November 10, 1983 and subsequently revised in letters dated January 21, 1986, February 13, 1987, March 9, 1987, April 14, 1987, and October 2, 1987.

This amendment revises the requirements of the Technical Specifications related to the definition of operability.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Carl Stahle

Carl Stahle, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 24 to
   License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Roger W. Kober
Rochester Gas and Electric Corporation

R. E. Ginna Nuclear Power Plant

cc:

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

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

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

Stanley B. Klimberg, Esquire
General Counsel
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Supervisor of the Town of Ontario
1850 Ridge Road
Ontario, New York 14519

Jay Dunkleberger
Division of Policy Analysis & Planning
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE DPR-18 - R. E. GINNA NUCLEAR POWER PLANT

DISTRIBUTION:
DOCKET 50-244
NRC PDR
LOCAL PDR
PDI-3 R/F
S. Varga
B. Boger
M. Rushbrook
C. Stahle
OGC-Bethesda
D. Hagen
E. Jordan
J. Partlow
T. Barnhart (4)
Wanda Jones
E. Butcher
Tech. Review Branch: E. Conner, Region I
ACRS (10)
GPA/PA
ARM/LFMB
D. Haverkamp, Region I

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-18

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

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated November 10, 1983 as supplemented by letters, dated January 21, 1986, February 13, 1987, March 9, 1987, April 14, 1987, and October 2, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 24 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Vernon L. Rooney, Acting Director
Project Directorate I-8
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 27 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 24

FACILTIY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace Appendix "A" Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
i	i
1-1	1-1
N/A	1-1a*
N/A	3.0-1*
N/A	3.0.2*
N/A	3.0.3*
N/A	3.0.4*
3.1-1 through 3.1-3	3.1-1 through 3.1-3
3.1-5 through 3.1-6	3.1-5 through 3.1-6
3.1.6a	3.1-6a**
3.1-25 through 3.1-28	3.1-25 through 3.1-28
3.1-29	3.1-29**
3.1-30	3.1-30**
3.1-31 through 3.1-33	3.1-31 through 3.1-33
3.2-1 through 3.2-3	3.2-1 through 3.2-3
3.3-1 through 3.3-8	3.3-1 through 3.3-8
3.3-13	3.3-13 through 3.3-14*
3.4-1 through 3.4-3	3.4-1 through 3.4-3**
3.5-1 through 3.5-4	3.5-1 through 3.5-4**
3.5-5 through 3.5-22	3.5-5 through 3.5-23**
3.10-1 through 3.10-2	3.10-1 through 3.10-2
3.16-9	3.16-9

*denotes new page
**No text change, repositioned on page

TABLE OF CONTENTS

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

1.1 Thermal Power

The rate that the thermal energy generated by the fuel is accumulated by the coolant as it passes through the reactor vessel.

1.2 Reactor Operating Modes

Mode	Reactivity $\Delta k/k\%$	Coolant Temperature (°F)
Refueling	≤ -5	$T_{avg} \leq 140$
Cold Shutdown	≤ -1	$T_{avg} \leq 200$
Hot Shutdown	≤ -1	$T_{avg} \geq 540$
Operating	> -1	$T_{avg} \sim 580$

1.3 Refueling

Any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted.

1.4 Operable-Operability

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal or emergency electrical power sources (subject to Section 3.0.2), cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

3.0 LIMITING CONDITIONS FOR OPERATION

Applicability

3.0.1 In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied **because of circumstances in excess of those addressed** in the specification, within 1 hour action shall be initiated to place the unit in at least hot shutdown within the next 6 hours (i.e., a total of seven hours), and in at least cold shutdown within the following 30 hours (i.e., a total of 37 hours) unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a mode in which the specification is not applicable. If the action statement corresponding to the Limiting Condition for Operation that was exceeded contains time limits to hot and cold shutdown that are less than those specified above, these more limiting time limits shall be applied. Exceptions to these requirements shall be stated in the individual specifications.

3.0.2 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for

Operation, provided: (1) its corresponding normal or emergency power source is operable; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 1 hour, the unit shall be placed in at least hot shutdown within the next 6 hours, and in at least cold shutdown within the following 30 hours. This specification is not applicable in cold shutdown or refueling modes.

Basis

Specification 3.0.1 delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.3.2 requires two Containment Spray Pumps to be operable and provides explicit action requirements if one spray pump is inoperable. Under the terms of Specification 3.0.1, if both of the required Containment Spray Pumps are inoperable, the unit is required to be in at least hot shutdown within the following 6 hours and in at least cold shutdown in the next 30 hours. These time limits apply because the time limits for one spray pump inoperable (6 hours to hot shutdown, wait 48 hours then 30 hours to cold shutdown) are less limiting. As a further example, Specification 3.3.1 requires each Reactor Coolant System accumulator to be operable and provides explicit action requirements if one accumulator is inoperable. Under the terms of Specification 3.0.1, if more than one accumulator is inoperable, within 1 hour action shall be initiated to place the unit in at least hot shutdown within 6 hours and cold shutdown within an additional 6 hours. The time limit of 6 hours to hot shutdown and 30 hours to cold shutdown do not apply because the time limits for 1 accumulator inoperable are more limiting. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate action statement.

Specification 3.0.2 delineates what additional conditions must be satisfied to permit operation to continue, consistent with the action statements for power sources, when a normal or emergency power source is not operable. It allows operation to be governed by the time limits of the action statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual action statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.1.d requires in part that two emergency diesel generators be operable. The action statement provides for a maximum out-of-service time when one emergency diesel generator is not operable. If the definition of operable were applied without consideration of Specification 3.0.2, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.2 permit the time limits for continued operation to be consistent with the action statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be operable, and all redundant systems, subsystems, trains, components, and devices must be operable, or otherwise satisfy Specification 3.0.2 (i.e., be capable of performing their design function and have at least one normal or one emergency power source operable). If they are not satisfied, shutdown is required in accordance with this specification.

3.1 Reactor Coolant System

Applicability:

Applies to the operating status of the Reactor Coolant System when fuel is in the reactor.

Objective:

To specify those conditions of the Reactor Coolant System which must be met to assure safe reactor operation.

Specification:

3.1.1 Operational Components

3.1.1.1 Reactor Coolant Loops

a. When the reactor power is above 130 MWT (8.5%), both reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation.

b. If the conditions of 3.1.1.1.a are not met, then immediate power reduction shall be initiated under administrative control. If the shutdown margin meets the one loop requirements of Figure 3.10-2, then the power shall be reduced to less than 130 MWT. If the one loop shutdown margin of Figure 3.10-2 is not met, the plant shall be taken to the hot shutdown condition and the one loop shutdown margin shall be met.

c. Except for special tests, when the RCS temperature is at or above 350°F with the reactor power less than or equal to 130 MWT (8.5%), at least one reactor coolant loop and its associated steam generator and reactor coolant pump shall be in operation. The other loop and its associated steam generator must be operable so that heat could be removed via natural circulation. However, both reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

d. If the conditions of 3.1.1.1.c are not met, then

   (i) if one loop is in operation, but the other loop is not operable, restore the inoperable loop to operable status within 72 hours or take the plant to the hot shutdown condition and reduce the RCS temperature to less than 350°F within the next 12 hours, or

   (ii) if neither loop is in operation suspend all operations involving a reduction in boron concentration in the Reactor Coolant System and immediately initiate corrective action to return a coolant loop to operation.

e. When the RCS temperature is less than 350°F, at least two of the following coolant loops shall be operable:

   (i) reactor coolant loop A and its associated steam generator and reactor coolant pump.

   (ii) reactor coolant loop B and its associated steam generator and reactor coolant pump.

(iii) residual heat removal loop A.*

(iv) residual heat removal loop B.*

f. Except during steam generator crevice cleaning operations, at least one of the coolant loops listed in paragraph 3.1.1.1.e shall be in operation while RCS temperature is less than 350°F. However, both reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

g. If the conditions of 3.1.1.1.e are not met, immediately initiate corrective action to return the required loops to operable status, and if not in cold shutdown already, be in cold shutdown within 24 hours.

h. If the conditions of 3.1.1.1.f are not met, then suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

* The normal or emergency power source may be inoperable while in cold shutdown.

c. Whenever the reactor is at or above an RCS temperature of 350°F, both pressurizer code safety valves shall be operable with a lift setting of 2485 psig ±1%.

d. If one pressurizer code safety valve is not operable while the reactor is at or above an RCS temperature of 350°F, then either restore the inoperable valve to operable status within 15 minutes or be in at least hot shutdown within 6 hours and below an RCS temperature of 350°F within an additional 6 hours.

3.1.1.4 <u>Relief Valves</u>

a. Both pressurizer power operated relief valves (PORVs) and their associated block valves shall be operable whenever the reactor is at or above an RCS temperature of 350°F, or

(i) with one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and below an RCS temperature of 350°F within the following 6 hours, or

(ii) with one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s) and remove power from the block valve(s);

otherwise, be in at least hot shutdown within the next 6 hours and below an RCS temperature of 350°F within the following 6 hours.

3.1.1.5 Pressurizer

a. Whenever the reactor is at or above an RCS temperature of 350°F the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hrs. or have the reactor below an RCS temperature of 350°F and the RHR system in operation within an additional 6 hrs.

b. This requirement shall not apply during performance of RCS hydro test provided the test is completed and the pressurizer is operable per 3.1.1.5a within 16 hours.

3.1.1.6 Reactor Coolant System Vents

a. When the reactor is at hot shutdown or critical, at least one reactor coolant system vent path consisting of two valves in series shall be operable and closed* at each of the following locations:

1. Reactor Vessel head
2. Pressurizer steam space

*The PORV block valve is not required to be closed but must be operable if the PORV is capable of being opened.

b. With one or more vents at the above reactor coolant system vent path locations inoperable, startup may commence and/or power operation may continue provided at least one vent path is operable and the inoperable vent paths are maintained closed with motive power removed from the valve actuator of all the valves in the inoperable vent paths. If the requirements of 3.1.1.6a are not met within 30 days, be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

c. With all of the above reactor coolant system vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 72 hours or be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

Bases

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal

3.1.5 RCS Leakage

3.1.5.1 Detection Systems

3.1.5.1.1 With an RCS temperature greater than 350°F, two of the following leak detection systems, including one system sensitive to radioactivity, shall be in operation.

a. The containment air particulate monitor

b. The containment radiogas monitor

c. The containment atmosphere humidity detector

d. The containment water inventory monitoring system

3.1.5.1.2 When a system sensitive to radioactivity is not operable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once every 24 hours. Otherwise be in hot shutdown within the next 6 hours and have the RCS temperature less than 350°F within the following 6 hours.

3.1.5.2 RCS Leakage Limits

3.1.5.2.1 With the RCS temperature at or above 350°F, RCS leakage shall be limited to:

a. No leakage, if known to be through an RCS pressure boundary such as a pipe, vessel or valve body,

b. 10 gpm from a known leakage source other than the above,

c. 1 gpm from an unidentified leakage source,

d. .1 gpm tube leakage in one steam generator when averaged over 24 hours.

3.1.5.2.2 If the limits specified above are exceeded the following action is required.

a. With any RCS pressure boundary leakage, as defined in 3.1.5.2.1.a, be at hot shutdown within 6 hours and at an RCS temperature less than 350°F in the following 6 hours.

b. With leakage in excess of 3.1.5.2.1 b or c, reduce leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

c. With steam generator tube leakage in excess of 3.1.5.2.1d, be at hot shutdown within 6 hours and at an RCS temperature less than 350°F within the following 6 hours. If more than six months have elapsed since the last steam generator inspection, perform an inspection in accordance with the requirements of Technical Specification 4.2.

Basis

Water inventory balances, monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether or not it is from the reactor coolant system pressure boundary, can be a serious problem with respect to in-plant radioactivity contamination or it could develop into a still more serious problem if the leakage rate is of sufficient magnitude to effect cooling of the reactor core; and, therefore, first indications of such leakage should be investigated as soon as practicable.

If leakage is to the containment, its presence may be indicated by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. The rate of leakage to which the instrument is sensitive is 0.013 gpm within twenty minutes, assuming the presence of corrosion product activity.

b. The containment radiogas monitor is less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to greater than 10 gpm.

c. The humidity detector provides a backup to a. and b. The sensitivity range of this instrument is from approximately 2 gpm to 10 gpm.

d. A leakage detection system which determines leakage from water and steam systems within the containment collects and measures moisture condensed from the containment atmosphere by cooling coils of the main recirculation units. This system provides a dependable and accurate means of determining total leakage, including leaks from the cooling coils. This system can detect leakage from approximately 1/2 gpm to 10 gpm.

Indication of leakage from the above sources should be cause for an investigation and could require a containment entry and limited inspection at power of the reactor coolant system. Visual inspection procedures, i.e., looking for steam, floor wetness or boric acid crystalline formations, would be used.

It should be noted that detection systems sensitive to radioactivity will have an indication that is sensitive to the coolant activity and the location of the leak as well as the leak rate. Also since leakage directly into the containment could be from a variety of sources, such as the component cooling system, the service water system, the secondary system, the reactor make-up water system, the chemical and volume control system, the seal injection system, the sampling system, as well as the primary coolant system, an increase in containment air moisture or sump actuation does not necessarily mean a primary system leak. Water inventory balances, liquid waste activities and tritium content can all be used in determining the nature of a leak inside the containment.

If leakage is to another system, it will be detected by the plant radiation monitors and/or water inventory control.

When the source of leakage has been investigated, the situation can be evaluated to determine if operation can be continued safely. This evaluation will be within the criteria of this specification.

a. A leak of any magnitude in a pipe, vessel, or valve body in the coolant system pressure boundary compromises the integrity of that system and significantly alters the probability of a loss-of-coolant accident occurring. Therefore, prompt shutdown of the reactor or isolation of the leaking component is required to reduce the consequences of this event or prevent its occurrence.

b. The identified leakage rate is restricted to less than 25% of the coolant make-up capability with the minimum charging capacity powered by emergency power. This does allow for further degradation of the system during the evaluation and shutdown process with assurance that adequate cooling make-up capability exists. If the maximum allowable coolant activity existed, the 10 gpm leak rate would not result in doses in excess of the annual average allowed by 10 CFR Part 20.

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

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

Reference:

(1) FSAR Section 11.2.3, 14.2.4

3.1.6 <u>Maximum Reactor Coolant Oxygen, Fluoride, and Chloride Concentration</u>

3.1.6.1 With an RCS temperature above 200°F, the RCS chemistry shall be maintained within the following limits.

Contaminant	Steady-State Limit (ppm)	Transient Limit (ppm)
*Oxygen	0.10	1.00
Chloride	0.15	1.50
Fluoride	0.15	1.50

3.1.6.2 With any one or more of the chemistry parameters in excess of its Steady State Limit, but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least hot shutdown within 6 hours and below an RCS temperature of 200°F within the following 30 hours.

3.1.6.3 With any one or more of the chemistry parameters in excess of its Transient Limit, be in at least hot shutdown within 6 hours and below an RCS temperature of 200°F within the following 30 hours and perform an engineering evaluation in accordance with 3.1.6.5.

3.1.6.4 With the RCS temperature at or below 200°F, the RCS chemistry shall be maintained within the following limits.

Contaminant	Normal Limit (ppm)	Transient Limit (ppm)
Oxygen	Saturated	Saturated

* Limits for Oxygen not applicable below 250°F.

Chloride	0.15	1.50
Fluoride	0.15	1.50

3.1.6.5 If the concentration of chloride or fluoride exceeds the Steady State Limit for more than 48 hours, or exceeds the Transient Limit, maintain the RCS pressure less than 500 psig and perform an engineering evaluation of the effects of the out of limit conditions on the structural integrity of the RCS. This evaluation shall determine that the RCS remains acceptable for continued operation prior to increasing RCS temperature and pressure above 200°F and 500 psig respectively.

Basis:

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the normal limits as specified, the integrity of the Reactor Coolant System is assured under all operating conditions (1). If normal limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin, the addition of hydrazine during subcritical operation, or adjustment of the hydrogen concentration in the volume control tank (2) during power operation. Because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the period of 24 hours for corrective action to restore concentrations within the limits has been established. If the corrective action has not been effective at the end of the

24 hour period, then the RCS will be brought below 200°F and the corrective action will continue. The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit a steady state concentration in excess of limit to exist for a longer period of time at the colder RCS temperatures and still provide the assurance that the integrity of the primary coolant system will be maintained. In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

<u>Reference</u>:

(1) FSAR, Section 4.2

(2) FSAR, Section 9.2

3.2 Chemical and Volume Control System

Applicability

Applies to the operational status of the chemical and volume control system.

Objective

To define those conditions of the chemical and volume control system necessary to assure safe reactor operation.

Specification

3.2.1 When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection. The minimum capability for boric acid injection shall be equivalent to that supplied from the refueling water storage tank.

3.2.2 The reactor shall not be taken above cold shutdown unless the following Chemical and Volume Control System conditions are met.

a. At least two charging pumps shall be operable.

b. Both boric acid transfer pumps shall be operable.

c. The boric acid tanks together shall contain a minimum of 2,000 gallons of a 12% to 13% by weight boric acid solution at a temperature of at least 145°F.

d. System piping and valves shall be operable to the extent of establishing two flow paths from the boric acid tanks to the Reactor Coolant System and a flow path from the refueling water storage tank to the Reactor Coolant System.

e. Both channels of heat tracing shall be operable for the above flow paths.

3.2.3 The requirements of 3.2.2 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.2.2 within the time period specified below, the reactor shall be placed in the hot shutdown condition within 6 hours. If the requirements of 3.2.2 are not satisfied within an additional 48 hours the reactor shall be in cold shutdown within the next 30 hours.

a. If only one charging pump is operable, then restore the second pump to operable status within 24 hours.

b. One boric acid pump may be out of service provided the pump is restored to operable status within 24 hours.

c. One boric acid tank may be out of service provided a minimum of 2,000 gallons of a 12% to 13% by weight boric acid solution at a temperature of at least 145°F is contained in the operable tank and provided that the tank is restored to operable status within 24 hours.

d. If only one flow path from the boric acid tanks is operable, then restore the second flowpath to operable status within 24 hours.

e. One channel of heat tracing may be out of service provided it is restored to operable status within 24 hours.

3.2.4 Whenever the RCS temperature is greater than 200°F and is being cooled by the RHR system and the over-pressure protection system is not operable, at least one charging pump shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull-stop position.

Basis

The chemical and volume control system provides control of the reactor coolant system boron inventory.(1) This is normally accomplished by using either of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the refueling water storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents (12% concentration of boric acid) to the charging pumps).

3.3 **Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, and Charcoal/HEPA Filters**

Objective

To define those conditions for operation that are necessary:

(1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident, and (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

3.3.1 Safety Injection and Residual Heat Removal Systems

3.3.1.1 The reactor shall not be taken above the mode indicated unless the following conditions are met:

a. Above cold shutdown, the refueling water storage tank contains not less than 300,000 gallons of water, with a boron concentration of at least 2000 ppm.

b. Above a reactor coolant system pressure of 1600 psig, each accumulator is pressurized to at least 700 psig with an indicator level of at least 50% and a maximum of 82% with a boron concentration of at least 1800 ppm.

c. At or above a reactor coolant system pressure and temperature of 1600 psig and 350°F, except during performance of RCS hydro test, three safety injection pumps are operable.

d. At or above an RCS temperature of 350°F, two residual heat removal pumps are operable.

e. At or above an RCS temperature of 350°F, two residual heat removal heat exchangers are operable.

f. At the conditions required in a through e above, all valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

g. At or above an RCS temperature of 350°F, A.C. power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D, and refueling water storage tank delivery valve 856. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. D.C. control power shall be removed from refueling water storage tank delivery valves 896A and B with the valves open.

h. At or above an RCS temperature of 350°F, check valves 853A, 853B, 867A, 867B, 878G, and 878J shall be operable with less than 5.0 gpm leakage each. The leakage requirements of Technical Specification 3.1.5.2.1 are still applicable.

i. Above a reactor coolant system pressure of 1600 psig, A.C. power shall be removed from accumulator isolation valves 841 and 865 with the valves open.

3.3.1.2 If the conditions of 3.3.1.1a are not met, then be at hot shutdown within 6 hours and at cold shutdown within an additional 30 hours.

3.3.1.3 The requirements of 3.3.1.1b and 3.3.1.1i may be modified to allow one accumulator to be inoperable or isolated for up to one hour. If the accumulator is not operable or is still isolated after one hour, the reactor shall be placed in hot shutdown within the following 6 hours and below a RCS pressure of 1600 psig within an additional 6 hours.

3.3.1.4 The requirements of 3.3.1.1.c may be modified to allow one safety injection pump to be inoperable for up to 72 hours. If the pump is not operable after 72 hours, the reactor shall be placed in hot shutdown within 6 hours and at an RCS pressure and temperature less than 1600 psig and 350°F within an additional 6 hours.

3.3.1.5 The requirements of 3.3.1.1d through h may be modified to allow components to be inoperable at any one time. More than one component may be inoperable at any one time provided that one train of the ECCS is operable. If the requirements of 3.3.1.1d through h are not satisfied within the time period specified below, the reactor shall be placed in hot shutdown within 6 hours and at an RCS temperature less than 350°F in an additional 6 hours.

a. One residual heat removal pump may be out of service provided the pump is restored to operable status within 72 hours.

b. One residual heat removal heat exchanger may be out of service for a period of no more than 72 hours.

c. Any valve, interlock, or piping required for the functioning of one safety injection train and/or one low head safety injection train (RHR) may be inoperable provided repairs are completed within 72 hours (except as specified in e. below).

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

e. Those check valves specified in 3.3.1.1 h may be inoperable (greater than 5.0 gpm leakage) provided the inline MOVs are de-energized closed and repairs are completed within 12 hours.

3.3.1.6 Except during diesel generator load and safeguard sequence testing or when the vessel head is removed, or the steam generator primary system manway is open, no more than one safety injection pump shall be operable whenever the overpressure portection system is required to be operable.

3.3.1.6.1 Whenever only one safety injection pump may be operable by 3.3.1.6, at least two of the three safety injection pumps shall be demonstrated inoperable a minimum of once per twelve hours by verifying that the control switches are in the pull-stop position.

3.3.2 Containment Cooling and Iodine Removal

3.3.2.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

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

b. Both containment spray pumps are operable.

c. Four recirculation fan cooler units including the associated HEPA filter units with demisters are operable.

d. Both post accident charcoal filter units are operable.

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

3.3.2.2 The requirements of 3.3.2.1 may be modified to allow components to be inoperable at any one time provided that 1) the time limits and other requirements specified in a through f below are satisfied, and 2) at least 1 containment spray pump, 3 fan cooler units, 3 HEPA filter units with demisters, and 1 charcoal filter unit and all required valves and piping associated with these components are operable. If these requirements are not satisfied, the reactor shall be in hot shutdown within 6 hours. If the requirements are not satisfied within an additional 48 hours, be in cold shutdown within the next 30 hours.

a. One HEPA filter unit or demister and/or associated recirculation fan cooler may be inoperable for a period of no more than 7 days.

b. One containment spray pump may be inoperable provided the pump is restored to operable status within 3 days.

c. Any valve or piping in a system, required to function during accident conditions, may be inoperable provided it is restored to operable status within 72 hours.

d. One post accident charcoal filter unit and/or its associated fan cooler may be inoperable provided the unit is restored to operable status within 7 days.

e. The spray additive system may be inoperable for a period of no more than 3 days provided that both charcoal filter units are operable.

3.3.3 Component Cooling System

3.3.3.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

a. Both component cooling pumps are operable.

b. Both component cooling heat exchangers are operable.

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

3.3.3.2 The requirements of 3.3.3.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.3.1 within the time period specified, the reactor shall be in hot shutdown within the next 6 hours. If the requirements of 3.3.3.1 are not satisfied within an additional 48 hours, the reactor shall be in the cold shutdown condition within the following 30 hours. If neither component cooling water loop is operable, the reactor shall be maintained below a reactor coolant system temperature of 350°F instead of at cold shutdown and corrective action shall be initiated to restore a component cooling water loop to operable status as soon as possible.

a. One component cooling pump may be out of service provided the pump is restored to operable status within 24 hours.

b. One heat exchanger or other passive component may be out of service provided the system may still operate at 100% capacity and repairs are completed within 24 hours.

3.3.4 Service Water System

3.3.4.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

a. At least two service water pumps, one on bus 17 and one on bus 18, and one loop header are operable.

b. All valves, interlocks, and piping associated with the operation of two pumps are operable.

3.3.4.2 Any time that the conditions of 3.3.4.1 above cannot be met, the reactor shall be placed in hot shutdown within 6 hours and in cold shutdown within an additional 30 hours.

Amendment No. ~~11~~, 24

3.3.5 Control Room Emergency Air Treatment System

3.3.5.1 The RCS temperature shall not be at or above 350°F unless the control room emergency air treatment system is operable.

3.3.5.2 The requirements of 3.3.5.1 may be modified to allow the control room emergency air treatment system to be inoperable for a period of 48 hours. If the system is not made operable within those 48 hours, the reactor shall be placed in hot shutdown within the next 6 hours and the RCS temperature less than 350°F in an additional 12 hours.

Basis

The normal procedure for starting the reactor is, first to heat the

The facility has four service water pumps. Only one is needed during the injection phase, and two are required during the recirculation phase of a postulated loss-of-coolant accident.(8) The control room emergency **air treatment system is designed to filter the control** room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following the Design Basis Accident.(9) Reactor operation may continue for a limited time while repairs are being made to the air treatment system since it is unlikely that the system would be needed. Technical Specification 3.3.5 applies only to the equipment necessary to filter the control room atmosphere. Equipment necessary to initiate isolation of the control room is covered by another specification.

The limits for the accumulator pressure and volume assure the required amount of water injection during an accident, and are based on values used for the accident analyses. The indicated level of 50% corresponds to 1108 cubic feet of water in the accumulator and the indicated level of 82% corresponds to 1134 cubic feet. The limitation of no more than one safety injection pump to be operable and the surveillance requirement to verify that two safety injection pumps are inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

References

(1) FSAR Section 9.3

(2) FSAR Section 6.2

(3) FSAR Section 6.3

(4) FSAR Section 14.3.5

(5) FSAR Section 1.2

(6) FSAR Section 9.3

(7) FSAR Section 14.3

(8) FSAR Section 9.4

(9) FSAR Section 14.3.5

3.4 Turbine Cycle

Applicability

Applies to the operating status of turbine cycle.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core. The Standby Auxiliary Feedwater System provides additional assurance of capability to remove decay heat from the core should the Auxiliary Feedwater System be unavailable.

Specification

3.4.1 When the RCS temperature is at or above 350°F, the following conditions shall be met:

a. A minimum turbine cycle code approved steam-relieving capability of eight (8) main steam valves available (except for testing of the main steam safety valves).

b. Two motor driven auxiliary feedwater pumps and their associated flow paths (including backup supply from the Service Water System) must be operable.

c. The steam turbine driven auxiliary feedwater pump must be capable of being powered from an operable steam supply system, and the pump's associated flow path (including backup supply from the Service Water System) must be operable. The steam turbine driven auxiliary feedwater pump must be shown to be operable prior to exceeding 5% power.

d. A minimum of 22,500 gallons of water shall be available in the condensate storage tanks for the Auxiliary Feedwater System.

e. Two Standby Auxiliary Feedwater pumps and associated flow paths (including flow path from the Service Water System) must be operable.

3.4.2 Actions To Be Taken If Conditions of 3.4.1 Are Not Met

a. With one or more main steam code safety valves inoperable, restore the inoperable valve(s) to operable status within 4 hours or be in at least hot shutdown within the next 6 hours and at an RCS temperature below 350°F within the following 6 hours.

b. With one auxiliary feedwater pump or its flow path inoperable, restore the pump or flow path to operable status within 7 days. If the pump or flow path is not restored to operable status within 7 days submit a Special Report within an additional 30 days in accordance with Specification 6.9.2 outlining the cause of the inoperability and plans for restoring the pump or flow path to operable status.

c. With two auxiliary feedwater pumps or their flow paths inoperable, restore the inoperable pumps or flow paths to operable status within 72 hours or be in hot shutdown within the next 6 hours and at an RCS temperature below 350°F within the following 6 hours.

d. With one standby auxiliary feed pump or flow path inoperable, restore the inoperable pumps or flow paths to operable status within 7 days or be in hot shutdown within the next 6 hours and at an RCS temperature below 350°F within the following 6 hours.

e. With the required 22,500 gallons of water unavailable in the condensate storage tanks, within 4 hours, either:

   1. Restore the required amount of water, or

   2. Demonstrate the operability of the Service Water System as a backup supply to the auxiliary feed system and restore the required amount of water in the condensate storage tanks within 7 days, or

   3. Be in hot shutdown within the following 6 hours and at an RCS temperature of less than 350°F within the following 6 hours.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

The eight main steam safety valves have a total combined rated capability of 6,580,000 lbs/hr. This capability exceeds the total full power steam flow of 6,577,279 lbs/hr. In the event of complete loss of off-site electrical power to the station, decay heat removal is assured by either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves or atmospheric relief valves.(1) (2) The turbine driven pump can supply 200% of the required feedwater and one motor-driven auxiliary feedwater pump can supply 100% of the required feedwater for removal of decay heat from the plant, so any combination of two pumps can remove decay heat with a postulated single failure of one pump. The minimum amount of water in the condensate storage tanks is the amount needed to remove decay heat for 2 hours after reactor scram from full power.(4) An unlimited supply is available from the lake via either leg of the plant service water system for an indefinite time period.

The Standby Auxiliary Feedwater System is provided to give additional assurance of the capability to remove decay heat from the reactor. The system would be used only if none of the auxiliary feedwater pumps were available to perform their intended function. Since operability requirements are established for the auxiliary feedwater system, the Standby System would be required only if some unlikely event should disable all auxiliary feedwater pumps. The specified time to restore the Standby System to full capability is longer than for other components since the probability of being required to use the Standby System is extremely low.(3)

<u>References</u>:

(1) FSAR Section 10.4

(2) FSAR Section 14.1.9

(3) "Effects of High Energy Pipe Breaks Outside the Containment Building" submitted by letter dated November 1, 1973 from J. W. Amish, Rochester Gas and Electric Corporation to A. Giambusso, Deputy Director for Reactor Projects. U.S. Atomic Energy Commission

(4) L. D. White, Jr. letter to Mr. D. L. Ziemann, USNRC dated March 28, 1980

3.5 Instrumentation Systems

Objective

To delineate the conditions of the plant instrumentation and safety circuits.

Specification

3.5.1 Protection System Instrumentation

3.5.1.1 The Protection System Instrumentation shown on Table 3.5-1 shall be operable whenever the conditions specified in Column 6 are exceeded.

3.5.1.2 In the event the number of channels of a particular sub-system falls below the limits given in the columns 1 or 3 of Table 3.5-1, action shall be taken according to the requirements shown in column 5 of Table 3.5-1.

3.5.2 Engineered Safety Feature Actuation Instrumentation

3.5.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Table 3.5-2 shall be operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5-4, whenever the conditions specified in column 6 of Table 3.5-2 are exceeded.

3.5.2.2 In the event the number of channels of a particular subsystem falls below the limits given in columns 1 or 3 of Table 3.5-2, action shall be taken according to the requirements of column 5 of Table 3.5-2.

3.5.2.3 With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5-4, declare the channel inoperable and take action according to the requirements of column 5 of Table 3.5-2 until the channel is restored to operable status with the trip setpoint adjusted consistent with the Trip Setpoint value.

3.5.3 Accident Monitoring Instrumentation

3.5.3.1 The accident monitoring instrumentation channels shown in Table 3.5-3 shall be operable whenever the reactor is at or above hot shutdown.

3.5.3.2 When required by 3.5.3.1, with the number of operable accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5-3, either restore the inoperable channel(s) to operable status within 7 days, or be in at least hot shutdown within the next 12 hours.

3.5.3.3 When required by 3.5.3.1, with the number of operable accident monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table 3.5-3 either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

3.5.4 The radiation accident monitoring instrumentation channels shown in Table 3.5-6 shall be operable, whenever the reactor is at or above hot shutdown. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.5-6. Startup may commence or continue consistent with the action statement.

3.5.5 Radioactive Effluent Monitoring Instrumentation

3.5.5.1 The radioactive effluent monitoring instrumentation shown in Table 3.5-5 shall be operable at all times with alarm and/or trip setpoints set to insure that the limits of Specifications 3.9.1.1 and 3.9.2.1 are not exceeded. Alarm and/or trip setpoints shall be established in accordance with calculational methods set forth in the Offsite Dose Calculation Manual.

3.5.5.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

(i) the setpoint shall be immediately corrected without declaring the channel inoperable; <u>or</u>

(ii) immediately suspend the release of effluents monitored by the effected channel; <u>or</u>

(iii) declare the channel inoperable.

3.5.5.3 If the number of channels which are operable is found to be less than required, take the action shown in Table 3.5-5.

3.5.6 Control Room HVAC Detection Systems

3.5.6.1 During all modes of plant operation, detection systems for chlorine gas, ammonia gas and radioactivity in the control room HVAC intake shall be operable with setpoints to isolate air intake adjusted as follows:

chlorine, $\leq$ 5 ppm
ammonia, $\leq$ 35 mg/m^3
radioactivity, particulate $\leq 1 \times 10^{-8}$ μCi/cc
iodine $\leq 9 \times 10^{-9}$ μCi/cc
noble gas $\leq 1 \times 10^{-5}$ μCi/cc

3.5.6.2 With one of the detection systems inoperable, within 1 hour isolate the control room HVAC air intake. Maintain the air intake isolated except for short periods, not to exceed 1 hour a day, when fresh air makeup is allowed to improve the working environment in the control room.

Basis

During plant operations, the complete instrumentation system will normally be operable. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels inoperable since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is inoperable.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

The radioactive liquid effluent instrumentation is provided to monitor and/or control, as applicable, the releases of radioactive materials in liquid effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR Part 50.

Control room HVAC detection systems are designed to prevent the intake of chlorine, ammonia and radiation at concentrations which may prevent plant operators from performing their required functions. Concentrations which initiate isolation of the control room HVAC system have been established using the guidance of several established references (2-4).

The chlorine isolation setpoint is 1/3 of the toxicity limit of reference 2 but slightly greater than the short term exposure limit of reference 4. The ammonia setpoint is established at approximately 1/3 of the toxicity limit for anhydrous ammonia in reference 2 and equal to the short term exposure limit of reference 4. The setpoints for radioactivity correspond to the maximum permissible concentrations of reference 3 for Cs-137, I-131 and Kr-85.

References

1. Updated FSAR - Section 7.2.

2. USNRC Regulatory Guide 1.78, June 1974, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.

3. 10 CFR 20 Appendix B, Table I.

4. Threshold Limit Values for Chemical Substances and Physical Agents in the Work Environment, 1982. Published by American Conference of Governmental Industrial Hygienists.

TABLE 3.5-1
PROTECTION SYSTEM INSTRUMENTATION

NO.	FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
1.	Manual	2	1	2		1	when RCCA is withdrawn
2.	Nuclear Flux Power Range low setting	4	2	3	For low setting, 2 of 4 power range channels greater than 10% F.P.	2 Note 1	when RCCA is withdrawn
	high setting	4	2	3		2	when RCCA is withdrawn
3.	Nuclear Flux Intermediate Range	2	1	1	2 of 4 power range channels greater than 10% F.P.	3 Note 1	when RCCA is withdrawn
4.	Nuclear Flux Source Range	2	1	2	1 of 2 intermediate range channels greater than 10^{-10} amps.	4 Note 1	Note 2
		2	0	1		4	Note 3
5.	Overtemperature Δ T	4	2	3		2	Hot Shutdown
6.	Overpower Δ T	4	2	3		2	Hot Shutdown
7.	Low Pressurizer Pressure	4	2	3		2	5% power
8.	Hi Pressurizer Pressure	3	2	2		5	Hot Shutdown
9.	Pressurizer-Hi Water Level	3	2	2		5	5% power
10.	Low Flow in one loop (≥ 50% F.P.)	3/loop	2/loop (either loop)	2/loop (both loops)		5	5% power
	Low Flow both loops (8.5%-50% F.P.)	3/loop	2/loop (both loops)	2/loop (either loop)		6	5% power

TABLE 3.5-1 (Continued)
PROTECTION SYSTEM INSTRUMENTATION

NO. FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
11. Turbine Trip	3	2	2		5	50% Power
12. Steam Flow Feedwater flow mismatch with Lo Steam Generator Level	2 SF-FF and 2 SG level per loop	1 SF-FF coincident w/ 1 Lo SG level in same loop	2 SF-FF or 2 Lo SG level per loop		6	Hot Shutdown
13. Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop		5	Hot Shutdown
14. Undervoltage 4 KV Bus	2/bus	1/bus (both busses)	2/bus (on either bus)		6	5% Power
15. Underfrequency 4 KV Bus	2/bus	1/bus (both busses)	2/bus (on either bus)		6	5% Power
16. Quadrant power tilt monitor (upper & lower ex-core neutron detectors)	1	NA	1		Log individual upper & lower ion chamber currents once/hr & after a load change of 10% or after 48 steps of control rod motion	Hot Shutdown

TABLE 3.5-1 (Continued)
PROTECTION SYSTEM INSTRUMENTATION

NO.	FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
17.	Circulating Water Flood Protection						
	a. Condenser	2 sets of 3	2 of 3 in either set	2 of 3 in both sets		Power operation may be continued for a period of up to 7 days with 1 channel (1 set of three) inoperable or for a period of 24 hrs. with two channels (2 sets of of three) inoperable. Otherwise be in hot shutdown in an additional 6 hours.	Hot Shutdown
	b. Screenhouse	2 sets of 3	2 of 3 in either set	2 of 3 in both sets		Power operation may be continued for a period of up to 7 days with 1 channel (1 set of three) inoperable or for a period of 24 hrs. with two channels (2 sets of of three) inoperable. Otherwise be in hot shutdown in an additional 6 hours.	Hot Shutdown
18.	Loss of Voltage 480V Safeguards Bus	2 sets of 2/bus	1 of 2 in each set in one bus	2 of 2 in one of the two sets		7	T_{RCS} = 350°F

TABLE 3.5-1 (Continued)
PROTECTION SYSTEM INSTRUMENTATION

	1	2	3	4	5	6
NO. FUNCTIONAL UNIT	TOTAL NO. of CHANNELS	NO. of CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	CHANNEL OPERABLE ABOVE
19. Degraded Voltage 480V Safeguards Bus	2/bus	2/bus	1/bus		7	T_{RCS} = 350°F

NOTE 1: When block condition exists, maintain normal operation.
NOTE 2: Channels shall be operable at all modes below the bypass condition with the reactor trip system breakers in the closed position and control rod drive system capable of rod withdrawal.
NOTE 3: Channels shall be operable of all modes below the bypass condition except during refueling defined to be when fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.
F.P. = Full Power

TABLE 3.5-2
ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO. FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
1. SAFETY INJECTION						
a. Manual	2	1	2		8	T_{RCS} = 350°F
b. High Containment Pressure	3	2	2		9	T_{RCS} = 350°F
c. Steam Generator Low Steam Pressure/Loop	3	2	2	Primary pressure less than 2000 psig	9	T_{RCS} = 350°F
d. Pressurizer Low Pressure	3	2	2	Primary pressure less than 2000 psig	9	T_{RCS} = 350°F
2. CONTAINMENT SPRAY						
a. Manual	2	2**	2		10	Cold Shutdown
b. Hi-Hi Containment Pressure (Containment Spray)	2 sets of 3	2 of 3 in both sets	2 per set in either set		11	Cold Shutdown

** Must actuate 2 switches simultaneously.

TABLE 3.5-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO.	FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
3.	AUXILIARY FEEDWATER						
	Motor and Turbine Driven						
	a. Manual	1/pump	1/pump	1/pump		8	T_{RCS} = 350°F
	b. Stm. Gen. Water Level-low-low						
	i. Start Motor Driven Pumps	3/stm.gen.	2/stm.gen. either gen.	2/stm.gen. both gen.		9	T_{RCS} = 350°F
	ii. Start Turbine Driven Pump	3/stm.gen.	2/stm.gen. both gen.	2/stm.gen. either gen.		12	T_{RCS} = 350°F
	c. Loss of 4 KV Voltage Start Turbine Driven Pump	2/bus	1/bus (both buses)	2/bus (either bus)		12	T_{RCS} = 350°F
	d. Safety Injection Start Motor Driven Pumps		(see Item 1)				
	e. Trip of both Feedwater Pumps starts Motor Driven Pumps	2/pump	1/pump both pumps	2/pump either pump		6	5% power
	Standby Motor Driven						
	a. Manual	1/pump	1/pump	1/pump		8	T_{RCS} = 350°F

TABLE 3.5-2 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO.	FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP*	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
4.	CONTAINMENT ISOLATION						
4.1	Containment Isolation						
	a. Manual	2	1	2		10	Cold Shutdown
	b. Safety Injection (Auto Actuation)		(See Table 3.5-2, Item 1)				
4.2	Containment Ventilation Isolation						
	a. Manual	2	1	1		13	Cold Shutdown
	b. High Containment Radioactivity	2	1	2		13	Cold Shutdown
	c. Manual Spray		(See Table 3.5-2, Item 2a)				
	d. Safety Injection		(See Table 3.5-2, Item 1)				

TABLE 3.5-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO.	FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
5.	STEAM LINE ISOLATION						
	a. Hi-Hi Steam Flow with Safety Injection	2 Hi-Hi SF with S.I. for each loop	1 SF with S.I. in each loop	***		12	*T_{RCS} = 350°F w/MSIV's open
	b. Hi Steam Flow and 2 of 4 Low T_{AVG} with Safety Injection	2 Hi SF and 4 Low T_{avg} with S.I. for each loop	1 Hi SF and 2 Low T_{avg} with S.I. for each loop	***		12	*T_{RCS} = 350°F w/MSIV's open
	c. Containment Pressure	3	2	2		9	*T_{RCS} = 350°F w/MSIV's open
	d. Manual	1/loop	1/loop	1/loop		8	*T_{RCS} = 350°F w/MSIV's open
6.	FEEDWATER LINE ISOLATION						
	a. Safety Injection	(See Table 3.5-2, Item 1)					
	b. Hi Steam Generator Level	3/loop	2/loop in either loop	2/loop in both loops		9	**T_{RCS} = 350°F w/FW Isol valves open

* RCS temperature may be above 350°F if MSIV's are closed.
** RCS temperature may be above 350°F if FW Isol. valves are closed.
*** Both trains must be capable of providing a S.I. signal to each loop.

ACTION STATEMENTS

1. With the number of operable channels one less than the Minimum Operable Channels requirement, restore the inoperable channel to operable status within 48 hours or be in hot shutdown with all RCCA's fully inserted within the next 6 hours.

2. With the number of operable channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour and the requirements for the minimum number of channels operable are satisfied. However, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels.

   With the number of operable channels less than the Minimum Operable Channels requirement, be at a condition where operability is not required according to Column 6 of Table 3.5-1 within 6 hours.

3. With the number of operable channels one less than the Minimum Operable Channels requirement, suspend all operations involving positive reactivity changes and have all RCCA's fully inserted within 6 hours.

4. With the number of operable channels one less than the Minimum Operable Channels requirement, suspend all operations involving positive reactivity changes. If the channel is not restored to operable status within 48 hours, open the reactor trip breaker within the next hour.

5. With the number of operable channels one less than the Total Number of Channels, operation may proceed until the next Channel Functional Test provided the inoperable channel is placed in the tripped condition within 1 hour. With the number of operable channels one less than the Minimum Operable Channels requirement, or at the time of the next required Channel Functional Test referenced above, be at a condition where channel operability is not required according to Column 6 of Table 3.5-1 within the next 6 hours.

6. With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of a reactor trip signal, operation may proceed until this Channel Functional Test. At the time of this next Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels, be at a condtion where channel operability is not required according to Column 6 of Table 3.5-1 within the next 6 hours.

7. With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of a trip signal, operation may proceed until this Channel Functional Test. At the time of this Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels, either

    a) be at Hot Shutdown within the next 6 hours and an RCS temperature less than 350°F within the following 6 hours, or

    b) energize the affected bus with a diesel generator.

8. With the number of operable channels one less than the Minimum Operable Channels required, restore the inoperable channel to operable status within 48 hours or be in Hot Shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

9. With the number of operable channels one less than the Total Number of Channels required, operation may proceed until the next Channel Functional Test provided the inoperable channel is placed in the tripped position within 1 hour. At the next Channel Functional Test, or at any time the number of operable channels is less than the Minimum Operable Channels required, be at Hot Shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

10. With the number of operable channels one less than the Minimum Operable Channels required, restore the inoperable channel to operable status within 48 hours or be in Hot Shutdown within an additional 6 hours, and at cold shutdown within the following 30 hours.

11. With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of an actuation signal, operation may proceed until this Channel Functional Test. At the time of this Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels required, be at Hot Shutdown within 6 hours and at Cold Shutdown within the following 30 hours.

12. With the nubmer of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. Should the next Channel Functional Test require the bypass of an inoperable chanlel to avoid the generation of an actuation signal, operation may proceed until this Channel Functional Test. At the time of this Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels required, be at hot shutdown within 6 hours and at an RCS temperature less than 350°F within 6 hours.

13. With the number of operable channels less than the Minimum Operable Channels required, operation may continue provided the containment purge and exhaust valves are maintained closed.

Table 3.5-3
Accident Monitoring Instrumentation

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS (7)	MINIMUM CHANNELS OPERABLE (7)
1. Pressurizer Water Level (1)	2	1
2. Auxiliary Feedwater Flow Rate (2)(3)	2/steam generator	1/steam generator
3. **Steam Generator Water Level -** Wide Range (3)	**1/steam generator**	**1/steam generator**
4. Reactor Coolant System Subcooling Margin Monitor (4)	2	1
5. Pressurizer PORV Position Indicator (5)	2/Valve	1/Valve
6. PORV Block Valve Position Indicator (1)	1/Valve	0/Valve
7. Pressurizer Safety Valve Position Indicator (5)	2/Valve	1/Valve
8. Containment Pressure (8)	2	1
9. Containment Water Level (Narrow Range, Sump A)	1(6)	1(6)
10. Containment Water Level (Wide Range, Sump B)	2	1
11. Core-Exit Thermocouples	4/core quadrant	2/core quadrant

Notes

(1) Emergency power for pressurizer equipment, NUREG-0737, item II.G.1.

(2) Auxiliary feedwater system flow indication, NUREG-0737, item II.E.1.2.

(3) Only 2 out of the 3 indications (two steam generator auxiliary feedwater flow and one wide-range steam generator level) are required to be operable, NUREG-0737, item II.E.1.2.

(4) Instrumentation for detection of inadequate core cooling, NUREG-0737, item II.F.2.1.

(5) Direct indication of relief and safety valve position, NUREG-0737, item II.D.3. Two channels include a primary detector and RTD as the backup detector.

(6) Operation may continue with less than the minimum channels operable provided that the requirements of Technical Specification 3.1.5.1 are met.

(7) See Specification 3.5.3 for required action.

(8) Containment pressure monitor, NUREG-0737, item II.F.1.4.

TABLE 3.5-4
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES*
1. SAFETY INJECTION AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. High Containment Pressure	$\leq$ 4.0 psig	$\leq$ 5.0 psig
c. Low Pressurizer Pressure	$\geq$ 1723 psig	$\geq$ 1715 psig
d. Low Steam Line Pressure	$\geq$ 514 psig	$\geq$ 500 psig
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. High-High Containment Pressure	$\leq$ 28 psig	$\leq$ 30 psig
3. CONTAINMENT ISOLATION		
a. Containment Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation Logic	Not Applicable	Not Applicable
b. Containment Ventilation Isolation		
1. Manual	Not Applicable	Not Applicable
2. High Containment Radioactivity	Note 3	Not Applicable
3. From Safety Injection	Not Applicable	Not Applicable
4. Manual Spray	Not Applicable	Not Applicable

TABLE 3.5-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES*
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. High Containment Pressure	$\leq$ 18 psig	$\leq$ 20 psig
c. High Steam Flow, Coincident with Low T_{avg} and SI	dp corresponding to $\leq 0.49 \times 10^6$ lbs/hr at 755 psig $T_{avg} \geq 545°F$	dp corresponding to $\leq 0.55 \times 10^6$ lbs/hr at 755 psig $T_{avg} \geq 543°F$,
d. High-High Steam Line Flow Coincident with SI	dp corresponding $\leq 3.6 \times 10^6$ lbs/hr at 755 psig	dp corresponding to $\leq 3.7 \times 10^6$ lbs/hr at 755 psig
5. FEED WATER ISOLATION		
a. High Steam Generator Water Level	$\leq$ 67% of narrow range instrument span each steam generator	$\leq$ 68% of narrow range instrument span each steam generator
6. AUXILIARY FEEDWATER		
a. Low-Low Steam Generator Water Level	$\geq$ 17% of narrow range instrument span each steam generator	$\geq$ 16% of narrow range instrument span each steam generator. See Note 1.
b. From Safety Injection	N.A.	N.A.
c. Loss of 4 kV Voltage (Start TAFP)	62% of 4160 volts Note 2	Note 2
d. Feedwater Pump Breakers Open (start MAFP)	Not Applicable	Not Applicable

TABLE 3.5-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. LOSS OF VOLTAGE		
a. 480 V Safeguards Bus Under-voltage (Loss of Voltage)	see Figure 2.3-1	
b. 480 V Safeguards Bus Under-voltage (Degraded Voltage)	see Figure 2.3-1	
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, (block, unblock SI)	≦2000 psig	≦2000 psig

Note 1: A positive 11% error has been included in the setpoint to account for errors which may be introduced into the steam generator level measurement system at a containment temperature of 286°F as determined by an evaluation performed on temperature effects on level systems as required by IE Bulletin 79-21.

Note 2: This setpoint value is from inverse time curve for CVT relay (406C883) with tap setting of 82 volts and time dial setting of 1. Delay at 62% voltage is 3.6 seconds. The allowable values are ±5% of the trip setpoint.

Note 3: The trip setpoints for containment ventilation isolation while purging shall be established to correspond to the limits of 10 CFR Part 20 for unrestricted areas. The setpoints are determined procedurally in accordance with Technical Specification 3.9.2 by calculating effluent monitor count rate limits, which take into account appropriate factors for detector calibration, ventilation flow rate, and average site meteorology.

*Allowable Values are those values assumed in accident analysis.

TABLE 3.5-5
Radioactive Effluent Monitoring Instrumentation

	Minimum Channels Operable	Action
1. Gross Activity Monitors (Liquid)		
a. Liquid Radwaste (R-18)	1	1
b. Steam Generator Blowdown (R-19)	1*	2
c. Turbine Building Floor Drains (R-21)	1	3
d. High Conductivity Waste (R-22)	1	1
e. Containment Fan Coolers (R-16)	1	3
f. Spent Fuel Pool Heat Exchanger (R-20)	1	3
2. Plant Ventilation		
a. Noble Gas Activity (R-14) (Providing Alarm and Isolation of Gas Decay Tanks)	1	4
b. Particulate Sampler (R-13)	1	5
c. Iodine Sampler (R-10B or R-14A)***	1	5
3. Containment Purge Vent		
a. Noble Gas Activity (R-12)	1+	(see Table 3.5-2 & Action 13 thereto)
b. Particulate Sampler (R-11)	1+	(see Table 3.5-2 & Action 13 thereto)
c. Iodine Sampler (R-10A or R-12A)***	1+	5
4. Air Ejector Monitor (R-15 or R-15A)***	1**	6
5. Waste Gas System Oxygen Monitor	1	7

* Not required when Steam Generator Blowdown is being recycled (i.e. not released)
\+ Required only during containment purges
** Not required during Cold or Refueling Shutdown
***See Table 3.5-6

TABLE 3.5-5 (Continued)

Table Notation

Action 1 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases from the tank may continue for up to 14 days, provided that prior to initiating a release:

1. At least two independent samples of the tank's contents are analyzed, in accordance with Specification 4.12.1.1.a, and

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

Action 2 - When Steam Generator Blowdown is being released (not recycled) and the number of channels operable is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue for up to 31 days, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10-7 uCi/gram:

1. At least once per 8 hours when the concentration of the secondary coolant is > 0.01 uCi/gram dose equivalent I-131.

2. At least once per 24 hours when the concentration of the secondary coolant is ≤ 0.01 uCi/gram dose equivalent I-131.

Action 3 - If the number of operable channels is less than required by the Minimum Channels operable requirement, effluent releases via this pathway may continue for up to 31 days provided that at least once per 24 hours grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10-7 uCi/gm.

Action 4 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue for up to 31 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for isotopic activity within 24 hours or R14A is operable and readings are reviewed at least once per 8 hours.

TABLE 3.5-5 (Continued)

Table Notation

Action 5 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue for up to 31 days, provided samples are continuously collected as required by Table 4.12-2 Item I with auxiliary sampling equipment.

Action 6 - If the number of operable channels is less than required by the Minimum Channels Operable and the Secondary Activity is $\leq 1 \times 10^{-4}$ uCi/gm, effluent releases may continue via this pathway provided grab samples are analyzed for gross radioactivity (beta or gamma) at least once per 24 hours. If the secondary activity is greater than 1×10^{-4} uCi/gm, effluent releases via this pathway may continue for up to 31 days provided grab samples are taken every 8 hours and analyzed within 24 hours.

Action 7 - If the channel is inoperable, a sample of the gas from the in service gas decay tank shall be analyzed for oxygen content at least once every 4 hours.

Table 3.5-6

Radiation Accident Monitoring Instrumentation

Instrument	Minimum Channels Operable	Action
1. Containment Area (R-29 and R-30)	2	1
2. Noble Gas Effluent Monitors		
i. Plant Vent (R-14A)	1	1
ii. A Main Steam Line (R-31)	1	1
iii. B Main Steam Line (R-32)	1	1
iv. Containment Purge (R-12A)	1*	1
v. Air Ejector (R-15A)	1	1

Action Statements

Action 1 - With the number of operable channels less than required by the Minimum Channels Operable requirements, either restore the inoperable channel(s) to operable status within 7 days of the event, or prepare and submit a Special Report to the Commission within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

* Only when the shutdown purge flanges are removed.

3.10 <u>Control Rod and Power Distribution Limits</u>

<u>Applicability</u>

Applies to the operation of the control rods and power distribution limits.

<u>Objective</u>

To ensure (1) core subcriticality after a reactor trip, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

<u>Specification</u>

3.10.1 <u>Control Rod Insertion Limits</u>

3.10.1.1 When the reactor is subcritical prior to startup, the hot shutdown margin shall be at least that shown in Figure 3.10-2. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions (547°F) if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon or boron.

3.10.1.2 When the reactor is critical except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn (indicated position).

3.10.1.3 When the reactor is critical, except for physics tests and control rod exercises, each group of control rods shall be inserted no further than the limits shown by the lines on Figure 3.10-1 and moved sequentially with a 100 (±5) step (demand position) overlap between successive banks.

3.10.1.4 During control rod exercises indicated in Table 4.1-2, the insertion limits need not be observed but the Figure 3.10-2 must be observed.

3.10.1.5 During measurement of control rod worth and shutdown margin, the shutdown margin requirement, Specification 3.10.1.1, need not be observed provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion. Each full length control rod not fully inserted, that is, the rods available for trip insertion, shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position (indicated) within 24 hours prior to reducing the shutdown margin to less than the limits of Specification 3.10.1.1. The position of each full length rod not fully inserted, that is, available for trip insertion, shall be determined at least once per 2 hours.

TABLE 3.16-1 (CONTINUED)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway andor Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	1 control 3 indicator June thru October each of 3 farms	At least once per 15 days.	Gamma isotopic and I-131 analysis of each sample.
	1 control 1 indicator November thru May one of the farms	At least once per 31 days.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	4 control 4 indicator (Off shore at Ginna)	Twice during fishing season including at least four species.	Gamma isotopic analysis on edible portions of each sample.
c. Food Products	1 control 2 indicator (On site)	Annual at time of harvest. Sample from two of the following: 1. apples 2. cherries 3. grapes	Gamma isotopic analysis on edible portion of sample.
	1 control 2 indicator (On site garden or nearest offsite garden within 5 miles in the highest D/Q meteorological sector)	At time of harvest. One sample of: 1. broad leaf vegetation 2. other vegetable	Gamma isotopic analysis on edible portions of each sample.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated November 10, 1983, Rochester Gas and Electric (RG&E or the licensee) submitted an application to amend Facility Operating License No. DPR-18 for R. E. Ginna Nuclear Power Plant (GNPP). The proposed amendment would provide additional clarification of the term "operable" and provide consistency between limiting conditions for operation (LCO) and the associated action statements. These proposed Technical Specification (TS) changes were in direct response to an NRC generic letter, dated April 10, 1980. RG&E requested a delay in the incorporation of this request in their letter of August 25, 1980; this request and delay was acceptable as documented in an NRC letter of April 9, 1981.

By letter dated January 21, 1986, the TS changes proposed by the earlier application were revised in their entirety. This revision was necessary due to an interim issuance of TS pages related to the Systematic Evaluation Program (SEP) and reissuance of the existing TSs in conjunction with conversion of the operating license from provisional to full-term status, thus making the November 1983 application out of date.

Justification for the initially proposed TS changes were supplemented by the January 1986 submittal. Following the staff review of the January 1986 amendment request and in response to staff comments, minor changes and clarifications to the proposed TS were forwarded by licensee letters of February 13 and March 9, 1987. Also, by letter dated April 14, 1987 and October 2, 1987, the licensee clarified its request in both the November 10, 1983 and January 21, 1986 submittals regarding the deletion of the reference to part-length control rods on page 3.10.2 of the Technical Specification. Part-length control rods were removed from the plant pursuant to a 10 CFR 50.59 analysis. The January 1986, et al., changes provided clarification of the amendment request and did not alter the proposed action or affect the staff's initial determination published in the Federal Register on January 26, 1984.

2.0 DISCUSSION AND EVALUATION

The proposed TS revisions include changing the definition of "operable" to a definition similar to that in the Standard TS (STS). This change requires additional changes to some Limiting Conditions for Operation (LCO) and Action Statements. RG&E states that most of the changes are one of three types as follows:

1. The LCO was changed to require the system or component to be operable at a lower temperature or condition, i.e. operable at an RCS temperature greater than 350 F versus requiring the equipment to be operable in order to be critical or at hot shutdown.

2. The action statement was changed to reflect the change in the LCO, i.e. if the LCO required the component to be operable at an RCS temperature greater than 350 F, then to clear the LCO, RCS conditions must be brought to below 350 F.

3. Time limits to hot shutdown and cold shutdown or to an RCS temperature less than 350 F were added to action statements where required.

The licensee concluded that changes such as Case 1, requiring a component to be operable at a lower condition, is inherently more conservative. In changes such as Case 2, if a component is not required in the LCO to be operable until a certain mode of operation, there is no reason for the Action Statement to require that RCS condition be reduced to modes below that referenced in the LCO. Therefore, Action Statements were changed to be consistent with the LCO.

The staff evaluated each of the licensee three reasons (Cases) for making changes as follows:

1. This case is where the licensee proposes to require the system or component to be operable at a lower temperature or condition than that in the current TS. (In most cases the changes are from the critical condition or from hot shutdown (T-avg $\geq$ to 540 F) to a temperature of $<$ 350 F. All such changes were found conservative compared to the current TS, will reduce the consequences of analyzed accidents and are, therefore, acceptable.

2. This case is where the proposed change is for TS consistency, making the action statements consistent with LCO. (In most cases, these changes insert the 350 F temperature value into the action statement). Since the purpose is to make the TS consistent, not to change requirements, such changes are editorial and are, therefore, acceptable.

3. In many of the current TS requirements, no time limits are specified for a component to be inoperable or for the plant to reach hot shutdown, ≤ 350 F, or cold shutdown conditions. The licensee's proposed TS changes would impose 6 hours to shutdown (be in at least hot shutdown within 6 hours), 6 hours to reduce RCS temperature below 350 F hours, and 24 hours to cooldown the reactor using shutdown cooling (be in cold shutdown within 24 hours). Some proposed TSs combine these basic time intervals for clarity. Since the proposed TS changes specify time requirements to reach specified plant conditions, they improve specificity and inspectability. They are also consistent with the time limits usually provided for other facilities. Therefore, the staff finds Case 3 changes acceptable.

In addition to the licensee types of changes, the staff adds the following two cases.

4. Some proposed TS changes utilize the wording of the STS, issued by the staff on July 27, 1981. Since this TS wording has received prior staff review and approval, and the wording clarifies the TS requirements, such changes are acceptable.

5. During the review of TS pages containing proposed changes, the licensee or staff noted that a specification needed a minor improvement or wording correction. These issues were discussed with the licensee, agreement was reached, and the licensee submitted revised TS pages on February 13 and March 9, 1987 documenting the minor correction. These changes are editorial, improve the TSs, and are, therefore, acceptable.

The following table lists the proposed changes to the indicated TS sections and notes the applicable Case Number for acceptability.

TABLE OF TS SECTIONS BEING CHANGED

TS Section	Subject of TS Section	Case No.
1.2	Definition of Modes	5
1.4	Definition of Operable-operability	4
3.0	Limiting Conditions for Operation	4
3.1.1.1	Reactor Coolant Loops	2/3
3.1.1.3	Pressurizer Safety Valves	1
3.1.1.4	Pressurizer Power Operated Relief Valves	1
3.1.1.5	Pressurizer Operability	1
3.1.5	Primary Coolant Leak Detection	1/2/3
3.1.6	Primary Coolant Chemistry	2/5
3.2	CVCS Operability	1/2
3.2.3.a	Charging Pump	3/5
3.2.3.d	Boric Acid Tank	3/5
3.2.4	One Charging Pump Inoperability	2
3.3	Emergency Core Cooling System	2
3.3.1	Safety Injection and RHR Operability	1/2
3.3.1.2	Refueling Water Storage Tank	3
3.3.1.3	Accumulator	2
3.3.1.4	Safety Injection Pumps	2
3.3.1.5	Inoperable Components	3
3.3.2	Containment Cooling and Iodine Removal	1/3
3.3.3	Component Cooling	1/3
3.3.4	Service Water	1/3

TS Section	Subject of TS Section	Case No.
3.3.5	Control Room Air Treatment	1/3
3.4.1	Auxiliary Feedwater	1/2/3
3.5.1	Protection System Instrumentation	1-5
3.5.2	ESFA Instrumentation	1-5
3.10.1	Control Rod Insertion Limit	5
3.16	Radiological Environmental Monitoring	5

Based on the above, we find the proposed revision of Technical Specifications regarding operability and related changes, as revised, acceptable.

In its November 10, 1983 submittal, the licensee proposed deleting the statement in the Technical Specifications on part-length control rods because these rods were removed pursuant to 10 CFR 50.59 analysis in February 1979. The effect of these part-length control rods on power distribution and reactivity has not been evaluated as part of the reload safety evaluations since that date. The deletion of the reference to part-length control rods is reflected by this amendment.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves only changes in administrative procedures and requirements. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

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

5.0 ACKNOWLEDGEMENT

Principal Contributors: E. Conner and D. Haverkamp

Dated: OCT 27 1987