ATTACHMENT A

Revise the Technical Specification pages as follows:

Remove	Insert
3.1-5	3.1-5
-	3.1-5a (new)
-	3.1-5b (new)
-	3.1-5c (new)
3.1-6	3.1-6
3.1-6a	---
3.1-7	3.1-7
3.1-8	3.1-8
3.1-9	3.1-9
3.1-9a	3.1-9a
-	3.1-9b (new)
-	3.1-9c (new)
-	3.1-9d (new)
4.3-4	4.3-4
-	4.3-4a (new)
4.3-5	4.3-5
4.3-6	4.3-6

c. Whenever the reactor is above cold shutdown, 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 above cold shutdown, then either restore the inoperable valve to operable status within 15 minutes or be in at least hot shutdown within 6 hours and in at least cold shutdown within the following 30 hours.

Basis:

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal operations and anticipated transients. Heat transfer analyses show that reactor heat equivalent to 130 MWT (8.5%) can be removed by natural circulation alone. Therefore, operation with one operating reactor coolant loop while below 130 MWT provides adequate margin.

The specification permits an orderly reduction in power if a reactor coolant pump is lost during operation between 130 MWT and 50% of rated power.(1) Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than one which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase.

When the reactor coolant system average temperature is above 350°F, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require one loop be in operation and the other loop be capable of removing heat via natural circulation.

When the reactor coolant system average temperature is between 200°F and 350°F or while in cold shutdown, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to prevent a sudden increase in reactivity if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the small pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant. When the boron concentration of the reactor

coolant system is to be increased, the process must be uniform to prevent sudden reactivity increases in the reactor during subsequent startup of the reactor coolant pumps. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump is running while the change is taking place. Emergency boration without a reactor coolant pump in operation is not prohibited by this specification.

Prohibiting reactor coolant pump starts without a large void in the pressurizer or without a limited RCS temperature differential will prevent RCS overpressurization due to expansion of cooler RCS water as it enters a warmer steam generator. A 38% level in the pressurizer will accommodate the swell resulting from a reactor coolant pump start with an RCS temperature of 140°F and steam generator secondary side temperature of 340°F, or the maximum temperature which usually exists prior to cooling the reactor with the RHR system.

Temperature requirements for the steam generator correspond with measured NDT for the shell and allowable thermal stresses in the tube sheet.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve set point. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby

control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provides adequate defense against overpressurization.

References

(1) UFSAR Sections 7.2.2.4 and 7.2.2.2.11

3.1.1.4 Relief Valves (Requirements relocated to Specification 3.1.1.6)

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.

Basis:

Pressurizer

The requirement that 100 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain sufficient RCS pressure/subcooling to support natural circulation at hot shutdown and during cooldown.[1]

References

(1) Letter from L. D. White, Jr. to D. L. Ziemann, USNRC, dated October 17, 1979. Subject: Follow up actions resulting from the NRC staff reviews regarding the Three Mile Island Unit 2 Accident.

3.1.1.6 Reactor Coolant System Vents

Specification

a. When the reactor is at or above an RCS temperature of 350°F:

(1) Both Reactor Vessel head vent paths each consisting of two valves in series shall be operable and closed.

(2) Both Pressurizer Steam Space vent paths each consisting of a PORV and its associated block valve shall be operable.

Action

a. With an inoperable valve in one or both Reactor Vessel head vent path(s), within 1 hour close at least one valve in the inoperable path and remove motive power from its actuator and within the following 30 days close the remaining valve in the inoperable Reactor Vessel head vent path(s) with motive power removed from its valve actuator; 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.

b. With excessive seat leakage in one or both Pressurizer Steam Space vent path(s) PORV, within 1 hour close its (their) associated block valve(s) with power maintained to 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.

c. With one or both PORV(s) inoperable, within 1 hour either restore the PORV(s) to operable status or close its (their) associated block valve(s) and remove power to 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. If the PORV(s) is (are) not operable within 72 hours, prepare and submit a Special Report within 30 days outlining the cause and plans for restoring the PORV(s).

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to operable status or place its (their) associated PORV(s) switch(es) in manual control; 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.

e. With both vent paths at either the Reactor Vessel head or the Pressurizer Steam Space inoperable, continued operation is permitted provided at least one vent path at each location is operable within 30 days; otherwise be in hot shutdown within 6 hours and below 350°F within the following 6 hours.

f. With all the above reactor coolant system vent paths inoperable; 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 6 hours.

Basis:

Reactor Coolant System Vents

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head and one from the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become

inoperable. The electrical power for the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. A <u>safety-related</u> nitrogen accumulator system independent from the plant instrument air system is provided for each PORV as an alternate operating system and ensures operability of these valves.

The operability of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs with the nitrogen accumulator system to control reactor coolant system pressure. This function could be used for the steam generator tube rupture accident and for plant shutdown.

B. Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage as defined in Specification 3.1.5.2.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).

D. Manual control of a block valve to isolate a stuck-open PORV.

<u>Action a</u> requires closure of at least one valve in the inoperable Reactor Vessel head vent path with motive power removed from the valve actuator providing additional assurance precluding any inadvertent opening of the valves.

Action b includes the requirement to maintain power to closed block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control RCS pressure. Further, with one or both pressurizer steam space vent path(s) PORV(s) having excessive leakage but capable of being manually cycled, the PORV is considered to be operable. Excessive seat leakage is defined to be in excess of the limit defined in Specification 3.1.5.2.1b (i.e., 10 gpm from a known leakage source). Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage. However, the applicability requirements of the specification to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage so that maintenance can be performed on the PORVs to eliminate the seat leakage condition.

Action c includes the removal of power from a closed block valve as additional assurance to preclude any inadvertent opening of the block valve at a time in which the PORV may not be closed due to maintenance to restore it to operable status. Action c also permits continued operation for a period of time with the block valves closed to permit re-establishment of valve operability.

Action d establishes remedial measures that are consistent with the function of the block valves. If the block valve(s) cannot be restored to operable status within 1 hour, the remedial action is to place the PORV in manual control to preclude its automatic opening for an overpressure event to avoid the potential for a stuck-open PORV at a time that the block valve is open and inoperable. Manual control for the PORV is when the switch is placed in the closed position and still capable of being placed in the open position if required. This also allows manual operation of the PORV to depressurize the RCS if needed.

Action e permits continued operation for a period of time not to exceed 30 days provided at least one vent path at each location is operable or shutdown actions are required. The restoration time of 30 days is based on engineering judgement. This time is considered reasonable considering that restoration of one inoperable vent path ensures the capability exists to perform their intended safety function. The 30 day restoration time limit ensures the limiting condition for operation is met without reliance on the provisions of the action requirements.

Action f requires restoration of at least one of the vent paths to operable status within 72 hours or shutdown actions are required. When a pressurizer steam space vent path PORV is determined to be inoperable solely because of excessive seat leakage the vent path

may be considered operable. The restoration time of 72 hours is consistent with Specification 3.1.1.6c. Upon restoration of one of the vent paths, Specification 3.1.1.6e continues to apply from the point in time it is identified that a Limiting Condition for Operation is not met.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

4.3.4 <u>Reactor Coolant System Vents</u>

4.3.4.1 (Reference Surveillance Requirement 4.16.1b)

4.3.4.2 Except during cold and refueling shutdown each block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel unless the valve is closed in order to isolate an inoperable PORV.

4.3.4.3 The Nitrogen System for the PORVs shall be demonstrated operable at least once per 18 months by operating the PORVs through a complete cycle of full travel.

4.3.4.4 Each reactor coolant system vent path shall be demonstrated operable at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Verifying flow through the reactor coolant vent system vent paths using either liquid or gas.

4.3.4 Reactor Coolant System Vents

Basis for Surveillance Requirements

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Emergency (backup) power supplies are provided for both the PORVs and block valves. In addition, a nitrogen accumulator system independent from the plant instrument air system is provided for each PORV. The block valves have their motive power from safety related 480 volt AC busses and their control power from the safety related 125 volt DC system.

Surveillance Requirement 4.3.4.2 provides an exception for testing the block valves when they are closed to isolate an inoperable PORV. This precludes the need to cycle the valves when the PORV is inoperable or when maintenance is being performed to restore an inoperable PORV to operable status. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to operable status, i.e., completion of the ACTION statement fulfills the required surveillance requirement.

Surveillance Requirement 4.3.4.3 ensures operability of the Nitrogen System to perform its intended function independent of the plant instrument air system. The frequency for performing the surveillance is based on performing the testing during the unit refueling and maintenance outage.

Surveillance Requirement 4.3.4.4 ensures operability of each reactor coolant system vent path. The frequency for performing the surveillance is based on performing the testing during the unit refueling and maintenance outage.

4.3.5 Reactor Coolant Loops

4.3.5.1 When reactor power is above 130 MWt (8.5%), the reactor coolant pumps shall be verified in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.2 When the average coolant temperature is above 350°F but the reactor is not critical, when the reactor is at hot shutdown, or when the reactor is critical but reactor power is less than or equal to 130 MWt (8.5%):

a) the operating reactor coolant pump(s) shall be verified to be in operation and circulating reactor coolant at least once per 12 hours, and

b) if a reactor coolant pump is not operating, but must be operable, it shall be demonstrated operable once per 7 days by verifying correct breaker alignments and indicated power availability.

4.3.5.3 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F, and fuel is in the reactor, the following shall be performed to demonstrate a loop is operable. Tests need not be performed if a loop is not relied upon to satisfy the requirements of Specification 3.1.1.1.e.

a) to demonstrate a reactor coolant loop operable, the reactor coolant pump(s), if not in operation, shall be demonstrated operable at least once per 7 days by verifying correct breaker alignments and indicated power availability.

b. to demonstrate a residual heat removal pump is operable, the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a shall be performed.

4.3.5.4 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F and fuel is in the reactor, at least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.5 In addition to the above requirements, in order to demonstrate that a reactor coolant loop is operable, the steam generator water level shall be greater than or equal to 16% of the narrow range instrument span.

Basis:

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

The surveillance requirements on pressurizer equipment will assure proper performance of the pressurizer function and give early indication of malfunctions.

ATTACHMENT B

Introduction

Generic Letter 90-06 required RG&E to respond as to whether we intend to follow the staff positions included in Enclosure A and B of the generic letter, as applicable, or propose alternative measures and the proposed schedule for implementation. Enclosures A and B request that the actions identified in Section 3 of each enclosure be implemented. Generic Letter 90-06 specifically requested statements from RG&E addressing the following items:

1. Whether RG&E will commit to incorporate improvements 1, 2, and 3 of Section 3.1 of Enclosure A.

2. Specifically with respect to improvement 3 of Section 3.1 of Enclosure A, whether RG&E will commit to use those modified limiting conditions of operation for PORVs and block valves in the Technical Specifications for modes 1, 2, and 3 in Attachment A-1 of Enclosure A for Westinghouse designed plants with two PORVs.

3. Whether RG&E will submit a license amendment request to modify the Technical Specifications and commit to use the modified Technical Specifications for the low-temperature overpressure protection system concerning the limiting conditions of operation in modes 5 and 6 as identified in Attachment B-1 of Enclosure B to this generic letter for Westinghouse-designed plants.

4. For items 2 and 3 above, whether RG&E will submit modifications to our current Technical Specifications by the end of the first refueling outage that starts six months or later after the date of this letter (i.e., after December 25, 1990).

The following provides a summary of RG&E's response dated April 18, 1991.

1. RG&E presently has both the PORVs and block valves within the scope of Ginna's Quality Assurance Program. The PORVs and block valves, as described in Ginna Quality Assurance Manual, Appendix A are classified as safety-related.

2. The Quality Assurance Program elements, described in Generic Letter 90-06, are currently included within the scope of Ginna's plant operational Quality Assurance Program and Maintenance/refurbishment program for PORVs and block valves.

3. The specific improvement specified in the generic letter, is currently practiced at RG&E. Specifically, to procure replacement parts and spares and that complete components be procured in accordance with the original code requirements or a later code reconciled to the original requirements.

Therefore, RG&E has committed to the improvements 1, 2, and 3 of Section 3.1 of Enclosure A to the generic letter. With respect to requested statements for item 2 and 3 above, the model Technical Specification 3.4.4 of the generic letter for limiting conditions of operation for PORVs and block valves for modes 1, 2, and 3 were

essentially identical to Ginna Technical Specifications. The following exceptions and proposed changes are summarized below:

1. Ginna Technical Specifications do not address an operability requirement for PORV excessive seat leakage or a requirement to maintain power to the block valves as a result of an inoperable PORV. RG&E has proposed changes to address this specific guidance.

2. The specific guidance of the generic letter recommends, with respect to an inoperable PORV due to causes other than excessive seat leakage, that power be removed from the block valve. Initially RG&E planned, per our April 18, 1991 response to the generic letter, to require if the PORVs are inoperable due to causes other than seat leakage that the block valves be closed but power be maintained to the valve. Thus, if necessary, the operator could respond to a need for the PORVs more quickly without the additional actions necessary to restore power to the block valves. Because RG&E's definition for PORV inoperability is that it cannot be manually cycled; we agree that power should be removed from the PORV block value in this situation.

   The generic letter also requires that operability be restored within 72 hours or initiate shutdown actions. RG&E does not agree with the specific guidance recommended, since the PORVs are needed to perform specified functions during all reactor operating modes, from "Operating" to "Cold Shutdown"; it is not considered advantageous to change modes, thereby putting the plant in a transient condition. Our position for this concern is consistent with our initial response.

   RG&E plans to perform additional work on the PORVs/control circuits to meet EQ condition during the 1993 Refueling Outage. Therefore, RG&E has determined that upgraded (safety-related, environmentally qualified) PORVs provide an additional measure of safety, such that operability under conditions required for safety are assured. RG&E will concentrate on re-establishing valve operability, rather than initiate a shutdown procedure. We would, however, inform the NRC, by Special Report, if the valve is not restored to operable status within 72 hours along with our plans for restoration.

3. Ginna Technical Specifications do not address placing a PORV in manual control (i.e., switch in closed position, taken to open when valve is required to be opened) with one or both of its associated block valves inoperable - RG&E agrees that putting the PORV(s) in manual control is acceptable. RG&E is revising the Technical Specification accordingly. However, exceptions were taken to the shutdown provisions noted in the modified specifications of the generic letter.

4. With respect to the requested statement, item 3, concerning the limiting conditions of operation in modes 5 and 6 as identified in Attachment B-1 of Enclosure B of the Generic Letter, the following is provided:

Only one exception not currently addressed in Ginna Technical Specifications is noted. Model Technical Specifications 3.4.9.3b of the Generic Letter requires "With one PORV inoperable in Modes 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a ________ square inch vent within a total of 32 hours."

RG&E does not agree with this proposed change, because the proposed Technical Specification change from 7 days to 32 hours to establish a 1.1 in^2 vent with one PORV inoperable will limit Ginna's ability to clean up the RCS in preparation for a refueling outage.

In order to clean up the reactor coolant system (RCS), the system must remain pressurized ~300 psig so the Reactor Coolant Pump (RCP) can remain in service. The RCP is needed to mix the RCS when adding the hydrogen peroxide and must remain in service to provide a means to cleanup the RCS through the mixed bed demineralizers.

By forcing the establishment of a 1.1 in^2 vent within 32 hours, no cleanup of the RCS would be possible. This would significantly increase the exposure during the outage and increase the contamination problems inside containment.

In addition, it is questionable whether or not the H_2 level in the RCS will be less than the required 4 cc/kg. If the H_2 level can not be reduced below this level prior to opening of the 1.1 in^2 vent, an explosive mixture of H_2 in air may result once the vent is established.

These ALARA and safety concerns must be addressed prior to reducing the allowable time with one PORV inoperable when overpressure protection is required. Therefore, RG&E considers the present Technical Specification 3.15.1.1 acceptable as written.

5. With respect to the surveillance requirements proposed by the generic letter, Ginna Technical Specifications are analogous to the specific requirements proposed with only one exception. Ginna Technical Specifications do not address operability of the emergency (backup) power supply and demonstration of PORVs and block valves specifically in Ginna Technical Specifications. However, procedures are in place which ensure the specific guidance is met. RG&E has proposed a similar surveillance requirement consistent with the configuration of Ginna's emergency (backup) nitrogen (N_2) system for PORVs. Ginna does not utilize a manual transfer switchover to emergency sources for block valves. Operating the block valves through a complete cycle of full travel ensures operability of the block valve(s). The electrical power for the block valves is capable of being supplied from an emergency power source. Specification 4.6.1d3(b) requires, in part, verifying that the diesel generator starts from a normal standby condition and energizes the automatically connected emergency loads.

Existing Specification 4.3.4.2 requires, except during cold and refueling shutdown, each block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel unless the valve is already closed. Therefore, the specific guidance of the generic letter to perform a surveillance at least once per 18 months, will not be addressed. The existing Specification 4.3.4.2 satisfies the intent of the generic letter.

ADMINISTRATIVE CHANGES

Separation of existing Specification 3.1.1.6a into two distinct limiting conditions for operation to establish unique action requirements does not change the intent of the original specification. The proposed administrative change also ensures consistency with existing Specification 3.1.1.4a (proposed to be relocated).

Proposed action statement 3.1.1.6a will be included to address existing action requirements described in Specification 3.1.1.6b for Reactor Vessel head vent paths, except that (1) inoperability of both Reactor Coolant head vents will be addressed under proposed action statement 3.1.1.6e and (2) shutdown provisions will be changed from 30 hours to 6 hours to be below 350°F. The allowable time of 6 hours is consistent with other specifications and thereby considered to promote clarity. Hence, the proposed change to include action statement 3.1.1.6a is considered to be administrative in nature.

The proposed change to delete existing Specification Section 3.1.1.4 and relocating the provisions under proposed Specification Section 3.1.1.6 eliminates duplication of identical requirements under two individual specifications.

Existing Surveillance Requirement 4.3.5.6 will be renumbered to be 4.3.4.4 in order to group similar requirements. This administrative change also ensures consistency with proposed Specification 3.1.1.6.

Proposed action statement 3.1.1.6f is essentially unchanged. Deletion of the requirement to maintain the inoperable vent paths closed with power removed from the valve in the inoperable vent path, ensures consistency with other proposed changes. This requirement was relocated to proposed action statement 3.1.1.6a. Proposed action statement 3.1.1.6a address requirements for an inoperable Reactor Vessel head vent. Requirements for inoperable Pressurizer Steam Space vent paths are addressed separately. Hence, the proposed change to action statement 3.1.1.6f ensures consistency and is considered to be an administrative change.

Proposed Specification 3.1.1.3c will be revised to address conditions for operation above cold shutdown but less than an RCS temperature of 350°F. This proposed change to require both pressurizer code safety valves to be operable above cold shutdown ensures all operating conditions are being addressed. Existing Specification 3.1.1.3a describes conditions during cold shutdown or refueling. Proposed Specification 3.1.1.3d describes conditions for operation above cold shutdown. This proposed change is conservative and considered to be an administrative change.

Specification 3.1.1.3d will be revised to require with only one pressurizer code safety valve operable to return to cold shutdown conditions. This proposed change is also conservative and ensures consistency.

PORV operability is determined by manual actuation and not by performing a channel calibration. Surveillance Requirement (SR) 4.3.4.1 is considered inappropriate and will note reference to Surveillance Requirement 4.16.1b. This existing requirement duplicates SR 4.3.4.1. Operability of the PORVs is demonstrated by proposed SR 4.3.4.3. Operability of the block valves is demonstrated by proposed SR 4.3.4.2.

<u>10CFR50.92 EVALUATION</u>

The proposed change, in the Ginna Technical Specifications, does not involve a significant hazards consideration. The basis for this determination is as follows:

- There is no significant increase in the probability or consequences of an accident previously evaluated because the accident conditions and assumptions are not significantly affected by the proposed change.

  The proposed change to action statement 3.1.1.4a(i) [to be renumbered to be 3.1.1.6c] to include the removal of power from a closed block valve will provide additional assurance to preclude any inadvertent opening of the block valve at a time in which the PORV may not be operable to assure RCS integrity.

  The provision of the generic letter requires with one or both PORV(s) inoperable to initiate shutdown actions if PORV operability is not restored within 72 hours or 1 hour respectively. RG&E does not address these shutdown actions, but rather will concentrate on re-establishing valve operability. If the block valve(s) and power are not removed within 1 hour shutdown provisions must be initiated. These proposed requirement ensures that the PORVs will be available to perform their function prior to initiating a change in operational characteristics. Proposed Specification 3.1.1.6e provides allowances for continued operation for a maximum duration of 30 days. This provision is balanced by the potential for unavailability of the PORVs to perform their intended function prior to initiating a change in modes or other specified conditions. This proposed specification continues to parallel the existing requirements in Ginna Technical Specification.

  Proposed action statement 3.1.1.4a(ii) [to be renumbered to be 3.1.1.6d] includes a provision to place the block valves associated PORV(s) switch in manual control due to an inoperable block valve(s). This requirement precludes the automatic opening for an overpressure event to avoid the potential for a stuck-open PORV at a time that the block valve is open and inoperable. Manual control for the PORV places the switch in the closed position but the valve is still capable of being placed in the open position if required.

This allows manual operation of the PORV to depressurize the RCS if needed. The existing action statement does not include the requirement to place the PORV(s) in manual control due to an inoperable block valve. However, the risks associated with disabling the automatic function of the PORVs to open, credited in Ginna Chapter 15 analyses, by placing the PORVs in manual control will balance the risks for the potential for a stuck-open PORV at the time that the block valve is inoperable. Therefore, the PORV operability requirements are effectively unchanged by this proposed change.

The proposed change of maintaining power to closed block valves could potentially increase the probability of an inadvertent opening of a block valve. The safety impact is, however, not significant since the proposed changes are only applicable if the PORV is inoperable due to excessive seat leakage (Proposed action 3.1.1.6b). If the block valve were inadvertently opened, valve leakage would occur; however, the reactor coolant system would not undergo a rapid depressurization.

Proposed action statement 3.1.1.6b establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage and is therefore considered operable to perform its intended safety function. This proposed action statement is consistent with the guidance described in Generic Letter 90-09.

Proposed Surveillance Requirement 4.3.4.3 addresses operability of the Nitrogen System by demonstration of the PORVs at least once per 18 months by operating the PORVs through a complete cycle of full travel. This proposed addition was included as recommended by Generic Letter 90-06.

Based on the above efforts, the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

- The possibility of a new or different kind of accident from any previously evaluated is not created. In matters related to nuclear safety, all accidents continue to bound previous analyses. The proposed changes do not add or modify any equipment design nor do the proposed changes involve any significant operational changes to any plant systems.

The proposed change to place a PORV in manual control, as described in proposed action statement 3.1.1.6d, does not significantly alter operational capabilities for the PORV to perform its intended function. Placing the PORV in manual control is sufficient to preclude the potential for the PORV automatically opening for an over-pressure event and to avoid the potential for a stuck-open PORV at the time that the block valve is inoperable. Existing Specification 3.1.1.4a(ii) requires with one or more block valve(s) inoperable to remove power from the block valve(s) and allows power to be maintained to the PORVs with the automatic function available. Proposed Specification 3.1.1.6d requires the PORV to be placed in manual control and maintain power to the block valve. The

decision-making capability or response time of the operator to augment automatic functions of either the PORV or the block valve is essentially unchanged due to the proximity of their operational control features. The proposed change alleviates the possibility of an automatic opening of the PORV during restoration of the block valve to an operable status and therefore considered to be an improvement to the existing Specification 3.1.1.4a(ii).

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident previously evaluated.

- The proposed amendment does not involve a significant reduction in the margin of safety as defined in the basis for any technical specification because the results of the accident analyses which are documented in the UFSAR continue to bound operation under the proposed changes so that there is no safety margin reduction. The proposed amendment does not significantly change the requirements currently in Ginna Specifications but rather incorporates additional provisions considered following the TMI-2 accident, particularly with respect to required operator actions and equipment availability and performance. The proposed changes to Technical Specification limiting conditions for operations, action statements and surveillance requirements will enhance the margin of safety provided by the Technical Specifications. The increase in the margin of safety is provided by an increase in the probability that the PORVs would be available if needed for accident recovery or for low temperature overpressure protection. These proposed changes are considered to be an improvement to Ginna Technical Specifications.

  Therefore, the proposed changes do not involve a significant reduction in margin of safety.

CONCLUSION

On the basis of the above, RG&E has determined that the amendment request does not involve a significant hazards consideration.

Table 1

DETAILED DESCRIPTION OF CHANGES

PROPOSED PAGE	CHANGE	REASON FOR CHANGE
3.1-5, 3.1-5a through 3.1-5c	Basis relocated, FSAR references changed to UFSAR references	Administrative change to promote clarity
3.1-5	Change made requires pressurizer code safety valves to be operable above cold shutdown rather than 350°F.	To address all operating conditions above cold shutdown.
3.1-6	Provisions of Specification 3.1.1.4, Relief Valves, relocated under the provisions of Specification Section 3.1.1.6.	Relocated to eliminate duplication of requirements.
3.1-6	Existing Specification 3.1.1.5 basis section relocated to complement Specification. Additional text; was added to the basis.	Administrative change to promote clarity.
3.1-7	Relocated existing Specification 3.1.1.4a(i) and renumbered to be 3.1.1.6c.	Relocated to eliminate duplication of requirements and to provide consistency with recommendations of the generic letter.

PROPOSED PAGE	CHANGE	REASON FOR CHANGE
3.1-7	(1) Existing Specification 3.1.1.6a was revised to address specific requirements related to each type, i.e., Reactor Vessel head and Pressurizer Steam Space, of vent path.	1) Administrative change to promote clarity.
	(2) Note related to existing Specification 3.1.1.6a deleted.	(2) To provide consistency with the requirements of proposed Specification Section 3.1.1.6.
	(3) Revised Specification 3.1.1.6a operability requirement to provide clarification of the use of the term "operable", i.e., changed existing requirement from "hot shutdown or critical" to "at or above an RCS temperature of 350°F".	(3) To provide a consistent use of the term "operable" described in a previous Technical Specification Amendment (24). Consistent with existing Specification Section 3.1.1.4.
	(4) Included new action statement "a" to Specification 3.1.1.6 to close at least one valve in the inoperable Reactor Vessel head vent and remove power from the valve actuator within 1 hour and the remaining valve within 30 days otherwise implement shutdown requirements.	(4) To clearly identify action requirements related to the Reactor Vessel head vent from proposed actions related to PORVs, consistent with Generic Letter 90-06.
	(5) Included Specification 3.1.1.6b to address PORV requirements as a result of excessive seat leakage.	(5) To provide consistency with recommendations outlined in Generic letter 90-06.

PROPOSED PAGE	CHANGE	REASON FOR CHANGE
3.1-8	Included proposed action statement 3.1.1.6c (existing action statement 3.1.1.4a(i)) to address inoperable PORVs because of reasons other than excessive leakage, i.e., not capable of being manually cycled, and added requirement to remove power to the block valves and to submit a Special Report in 30 days if PORV(s) are not restored within 72 hours.	Included per recommendations of Generic Letter and RGE response to Generic Letter 90-06.
3.1-8	Revised existing action (ii) of Specification 3.1.1.4a to require that PORV(s) be placed in manual control if block valves are inoperable. Renumbered to be 3.1.1.6d.	Consistent with Generic Letter 90-06 recommendations.
3.1-8	Revised existing action statement "b" of Specification 3.1.1.6 (proposed action statement "e" of Specification 3.1.1.6) to address provisions for continued operation with inoperable vents for either the Reactor Vessel Head Vents or the Pressurizer Steam Space Vent Paths. Also changed time limit of 30 hours to 6 hours.	To provide consistency with proposed changes and to clearly separate the individual path locations as related to mode changes. Time limit change from 30 to 6 hours provides consistency with other specifications.
3.1-9	Deleted requirement from existing Specification 3.1.1.6c (renumbered to be 3.1.1.6f) to maintain the inoperable vent paths closed with power removed from the valve actuators of all valves in the inoperable vent paths. Revised time limit of 30 hours to 6 hours.	Proposed changes already address requirements individually for each of the path locations. Time limit change to provide consistency with time limits noted in other action statements.

PROPOSED PAGE	CHANGE	REASON FOR CHANGE
3.1-9, 3.1-9a through 3.1-9d	(6) Added basis section to address operability requirements for PORVs and block valves. Also included appropriate text from existing Specification Section 3.1.1.4 basis.	(6) To promote clarity. Also consistent with generic letter but site specific to Ginna.
4.3-4	(1) Included a new proposed Surveillance Requirement 4.3.4.3 which addresses the Nitrogen System. (2) Surveillance Requirement 4.3.4.2 includes additional text for clarification. (3) Existing Surveillance Requirement 4.3.5.6 renumbered to be 4.3.4.4 and relocated under the provisions of Surveillance Requirement Section 4.3.4, Relief Valves. (4) Deleted Surveillance Requirement 4.3.4.1.	(1) Satisfy RG&E April 18, 1991 commitment to provide changes in response to Generic Letter 90-06. (2) Administrative change to promote clarity. (3) To promote clarity by grouping similar surveillances together. (4) Operability of PORVs is demonstrated by proposed SR 4.3.4.3. Requirement will continue to be controlled administratively.
4.3-4a	Added site specific basis for Surveillance Requirement Section 4.3.4, Relief Valves.	Consistent with guidance provided in Generic Letter 90-06 except modified to be site specific to Ginna.
4.3-4a	Deleted Surveillance Requirement 4.3.4.1 and referenced SR 4.16.1b	SR inappropriate for section since PORV operability is determined by manual actuation. Referenced SR 4.16.1b since it duplicates requirement.
4.3-5	Existing Surveillance Requirement Section 4.3-4 relocated.	Administrative change to promote clarity.
4.3-6	Deleted existing Surveillance Requirement 4.3.5.6.	Renumbered to be 4.3.4.4 in order to group similar requirements in one location.