Attachment A

1. Remove Technical Specification pages 3.5-4, 3.5-4a, 3.10-3, 3.10-5, 3.10-6, 3.10-7, 3.10-8, 3.10-9, 3.10-10, 4.1-5, 4.1-6, 4.1-8, 4.1-10.

2. Insert the enclosed revised pages 3.5-4, 3.5-4a, 3.10-3, 3.10-5, 3.10-6, 3.10-7, 3.10-8, 3.10-9, 3.10-10, 3.10-14, 4.1-5, 4.1-6, 4.1-8, 4.1-10.

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSABLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 5 CANNOT BE MET
11.	Turbine Trip	3	2	2	1		Maintain 50% of rated power
12.	Steam Flow Feedwater flow mismatch with Lo Steam Generator Level	2/loop	1/loop	1/loop	1/loop		Maintain hot shutdown
13.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop		Maintain hot shutdown
14.	Undervoltage 4 KV Bus	2/bus	1/bus	1/bus	_*		Maintain hot shutdown
15.	Underfrequency 4 KV Bus	2/bus	1/bus (both busses)	1/bus	_*		Maintain hot shutdown
16.	Quadrant power tilt monitor (upper & lower ex-core neutron detectors)	1	_*	1 or Log individual upper & lower ion chamber currents once/hr & after a load change of 10% or after 30" of control rod motion	_*		Maintain hot shutdown

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NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSABLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 5 CANNOT BE MET
17.	Circulating Water Flood Protection						
	A. Screenhouse	2	1	2+	_*		Power operation may be continued for a period of up to 7 days with 1 channel inoperable or for a period of 24 hrs. with two channels inoperable.
	B. Condenser	2.	1	2+	_*		Power operation may be continued for a period of up to 7 days with 1 channel inoperable or for a period of 24 hrs. with two channels inoperable.

NOTE 1: When block condition exists, maintain normal operation

F.P. = Full Power
* Not Applicable
*** If a functional unit is operating with the minimum operable channels, the number of channels to trip the reactor will be column 3 less column 4.
+ A channel is considered operable with 1 out of 2 logic or 2 out of 3 logic.

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

a. Movable detectors

b. Core-exit thermocouples

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

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

$F_Q(Z) = 4.64*K(Z)$ for $P < .5$ 16

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

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

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

3.10.2.3 Except for physics tests, if the quadrant to average power tilt ratio, exceeds1.02 but is less than 1.12, 16

, then within two hours:

a. Correct the situation, or

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

c. Limit power to 75% of rated power.

3.10.3 Control Rod Drop Time

3.10.3.1 While critical, the individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 1.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 540°F, and

b. All reactor coolant pumps operating.

3.10.3.2 With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to criticality.

3.10.4 Control Rod Group Height

3.10.4.1 While critical, and except for physics testing, all full length (shutdown and control) rods shall be operable and positioned within ± 12 steps (indicated position) of their group step counter demand position.

3.10.4.2 With any full length rod inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untripable, determine that the shutdown margin requirement of Specification 3.10.1.1 is satisfied within 1 hour and be in hot shutdown within 6 hours.

3.10.4.3 With one full length rod inoperable due to causes other than addressed by 3.10.4.2, above, or misaligned from its group step counter demand position by more than ± 12 steps (indicated position), operation may continue provided that within one hour either:

3.10.4.3.1 The rod is restored to operable status within the above alignment requirements, or

3.10.4.3.2 The rod is declared inoperable and the shutdown margin requirement of Specification 3.10.1.1 is satisfied. Operations may then continue provided either:

a. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour, while maintaining the limit of Specification 3.10.1.3; or

b. The power level is reduced to less than or equal to 75% of rated power within the next one hour, and the high neutron flux trip setpoint is reduced to less than or equal to 85% rated power within the next

four hours (total of six hours) and the following evaluations are performed:

(i) The shutdown margin requirement of Specification 3.10.1.1 is determined at least once per 12 hours.

(ii) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

(iii) A reevaluation of each accident analysis of Table 3.10-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

c. If power has been restricted in accordance with (b), above, then following completion of the evaluation identified in (b), the power level and high neutron flux trip setpoint may be readjusted based on the results of the evaluation provided the shutdown margin requirement of Specification 3.10.1.1 is determined at least once per 12 hours.

3.10.4.4 With two or more full length rods inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in hot shutdown within 6 hours.

3.10.5 Control Rod Position Indication Systems

3.10.5.1 While critical, the analog rod position indication system and the step counters shall be operable and capable of determining the control rod positions within ± 12 steps.

3.10.5.2 With a maximum of one analog rod position indicator per bank inoperable either:

a. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

b. Reduce the power to less than 50% of rated power within 8 hours.

3.10.5.3 With a maximum of one step counter per bank inoperable either:

a. Verify that all analog rod position indicators for the affected bank are operable and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

b. Reduce the power to less than 50% of rated power within 8 hours.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis.(1) In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors.

The lines shown on Figure 3.10-1 meet the shutdown requirement. The maximum shutdown margin requirement occurs at end-of-cycle life and is based on the value used in analysis of the hypothetical steam break accident. Early in cycle life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to 1.9% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

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

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

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear

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

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

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 540°F and with both reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The various control rod banks (shutdown banks, control banks A, B, C, and D are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position.[2] These are known as the step counters and analog rod position indication, respectively.

Operability of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. The 12 step (7.5 inches) permissible indicated misalignment ensures that the 15 inch misalignment assumed in the safety analysis is met.

The action statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in power; either of these restrictions provide assurance of fuel rod integrity during continued operation. In

addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

References:

(1) Technical Supplement Accompanying Application to Increase Power - Section 14

(2) FSAR, Section 7.3

Table 3.10-1

ACCIDENT ANALYSIS REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE CONTROL ROD

Rod Insertion Characteristics

Rod Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Rod Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Steam Line Break

Rod Ejection

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range	S M*(3)	D(1) Q*(3)	B/W(2)(4) P(2)(5)	1) Heat balance calculation** 2) Signal to ΔT; bistable action (permissive, rod stop, trips) 3) Upper & lower chambers for axial offset** 4) High setpoint (≤ 109% of rated power) 5) Low setpoint (≤ 25% of rated power)
2. Nuclear Intermediate Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3. Nuclear Source Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4. Reactor Coolant Temperature	S	R	M(1)	1) Overtemperature-Delta T 2) Overpower - Delta T
5. Reactor Coolant Flow	S	R	M	
6. Pressurizer Water Level	S	R	M	
7. Pressurizer Pressure	S	R	M	
8. 4 Kv Voltage & Frequency	N.A	R	M	Reactor Protection circuits only
9. Analog Rod Position	S(1,2)	R	M	1) With step counters 2) Log analog rod positions each 4 hours when rod deviation monitor is out of service

* By means of the movable in-core detector system.

** Not required during hot, cold, or refueling shutdown but as soon as possible after return to power.

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TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Calibrate	Test	Remarks
10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With analog rod position 2) Log analog rod positions each 4 hours when rod deviation monitor is out of service
11. Steam Generator Level	S	R	M	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14. Boric Acid Tank Level	D	R	N.A.	Bubbler tube rodded weekly
15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
16. Volume Control Tank Level	N.A.	R	N.A.	
17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
18. Radiation Monitoring System	D	R	M	
19. Boric Acid Control	N.A.	R	N.A.	
20. Containment Drain Sump Level	N.A.	R	N.A.	
21. Valve Temperature Interlocks	N.A.	N.A.	R	
22. Pump-Valve Interlock	R	N.A.	N.A.	
23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block trip
24. Accumulator Level and Pressure	S	R	N.A.	

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TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency	FSAR Section Reference
1.	Reactor Coolant Samples	Gross Radioactivity Concentration (beta-gamma)	3 times/weekly and at least every third day (1) (7)	
		Radio-chemical (2)(4)	Monthly (6)	
		E Determination (2)	Monthly (6)	
		Tritium Concentration	Weekly (6)	
		Chloride and Fluoride	3 times/week and at least every third day	
		Oxygen	5 times/week and at least every second day except when below 250°F	
		Gross Radioiodine Concentration	Weekly (3) (6)	
2.	Reactor Coolant Boron	Boron concentration	Weekly	
3.	Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	
4.	Boric Acid Tank	Boron concentration	Twice/week	
5.	Control Rods	Rod drop times of all full length rods	After vessel head removal and at least once per 18 months (8)	7
6.	Full Length Control Rod	Movement of at least 10 steps in any one direction for any rod not fully inserted	Monthly	7
7.	Pressurizer Safety Valves	Set point	Each Refueling shutdown	4
8.	Main Steam Safety Valves	Set point	Each Refueling shutdown	10

10 μ Ci/gm from the previous measured level, the sampling frequency shall be increased to a minimum of once/day until a steady activity level is established.

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of E. A radiochemical analysis and calculation of E and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μ Ci/gm from the previous measured level.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μ Ci/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give equivalent I-131 activity.

(5) An isotopic analysis for DOSE EQUIVALENT I-131 concentration is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.

(6) Not required during a cold or refueling shutdown.

(7) During a cold or refueling shutdown, primary coolant Gross Radioactivity will be determined weekly.

(8) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.

Attachment B

By letter dated November 5, 1979, the NRC requested that the Technical Specifications for control rod misalignment be revised to be consistent with the Standard Technical Specification requirements. The proposed revision incorporates the NRC guidance into the Ginna Technical Specifications.

Attachment C

The proposed Amendment to the Provisional Operating License has been evaluated and determined to fall within the definition of Class III of 10 C.F.R. Section 170.22 thereby requiring a fee of $4,000.

The proposed amendment deals with a single safety issue, over-pressure protection of the reactor vessel. The issue has been clearly identified by an NRC position.