Attachment 2

To Letter from

D. C Hintz (WPSC) to H. R. Denton (NRC)

Dated August 1, 1986

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: Part 20, Section 30.34 of Part 30 Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensees are authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 68, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1, 3.1.2 and 3.1.4 through 3.1.28, of the Fire Protection Safety Evaluation Report. These modifications shall be completed by the dates specified in Table 3.1. Dates for resolution of items are specified in Table 3.2. In the event that these dates for completion cannot be met, the licensee shall submit a report explaining the circumstances and propose a revised schedule.

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heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.(1)

The curves of Figure TS 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.3 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.3 and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is assured is below these lines.

The curves are based on the following nuclear hot channel factors:

$$F^N_{\Delta H} = 1.55 \qquad F^N_Q = 2.51$$

and include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1 - P)]$$ where P is the fraction of rated power

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in Specification 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 insure that the DNBR is always greater at partial power than at full power.

4. Reactor Coolant Flow
   A. Low reactor coolant flow per loop $\geq$ 90% of normal indicated flow as measured by elbow taps.
   B. Reactor coolant pump motor breaker open
      1) Low frequency set point $\geq$ 57.5 Hz
      2) Low voltage set point $\geq$ 75% of normal voltage

5. Steam Generators

   Low-low steam generator water level $\geq$ 5% of narrow range instrument span.

6. Reactor Trip Interlocks

   Protective instrumentation settings for reactor trip interlocks shall be as follows:
   A. Above 10% of rated power, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.
   B. Above 10% of rated power, the single-loop loss-of-flow trip is made functional.

7. Other Trips

   Undervoltage $\geq$ 75% of normal voltage

   Turbine Trip

   Manual Trip

   Safety Injection Trip (Refer to table TS 3.5-1 for trip settings)

Steam Generators

The low-low steam generator water level reactor trip assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System. (5)

Reactor Trip Interlocks

Specified reactor trips are by-passed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed set points above which these trips are made functional assure their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed set points will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10 percent of power.

Table TS 3.5-1 lists the various parameters and their set points which initiate safety injection signals. A safety injection signal also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Sec. 3.5 of these specifications for details of SIS signals.

References

(1) FSAR Section 14.1.1
(2) USAR, Page 14.1-5
(3) FSAR Section 14.3.1
(4) FSAR Section 14.1.8
(5) FSAR Section 14.1.10
(6) WCAP-8092

c. Any one of the following conditions of inoperability may exist during the time intervals specified. The reactor shall be placed in the hot shutdown condition if operability is not restored within the time specified, and it shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.

1. ONE of the operable charging pumps may be removed from service provided two pumps are again operable within 24 hours.

2. ONE boric acid transfer pump may be out of service provided both pumps are operable within 24 hours.

3. ONE channel of heat tracing may be out of service provided it is restored to operable status within 48 hours.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with any one of the two boric acid transfer pumps. An alternate method of boration will be use of the charging pumps directly from the Refueling Water Storage Tank. A third method will be to use the safety injection pumps. There are two sources of borated water available for injection through 3 different paths.

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

(2) The charging pumps can take suction directly from the Refueling Water Storage Tank containing a concentration of 1950 ppm boron solution. Reference is made to Specification 3.3.b.1.A.

d. Component Cooling System

1. The reactor shall not be made or maintained critical unless the following conditions are satisfied, except for low power physics tests and except as provided by Specification 3.3.d.2.

   A. TWO component cooling water trains are operable with each train consisting of:

      1. ONE component cooling water pump
      2. ONE component cooling water heat exchanger
      3. An operable flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions.

2. During power operation or recovery from an inadvertent trip, ONE component cooling water train may be inoperable for a period of 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

   -Achieve Hot Standby within the next 6 hours.
   -Achieve Hot Shutdown within the following 6 hours.
   -Achieve Cold Shutdown within an additional 36 hours.

e. Service Water System

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by Specification 3.3.e.2.

   A. TWO service water trains are operable with each train consisting of:

b. If, when the reactor is above 350°F, any of the conditions of Specification 3.4.a cannot be met within 48 hours, and except for the conditions of 3.4.c, the reactor shall be shut down and cooled below 350°F using normal operating procedures.

c. When the reactor is above 350°F, one auxiliary feedwater pump may be inoperable provided the pump is restored to operable status within 72 hours, or the reactor shall be shutdown and cooled below 350°F using normal operating procedures.

d. Reactor Power shall not exceed 50% of rated power unless two of the three turbine overspeed protection systems are operable. If two or more of the turbine overspeed protection systems are inoperable, then maintain power less than 50% of rated power. When only two systems are operable, an individual system may be blocked for no longer than 4 hours to allow for testing.

The secondary coolant activity is based on a postulated release of the contents of one steam generator to the atmosphere. This could happen, for example, as a result of a steam break accident combined with failure of a steam line isolation valve. The limiting dose for this case results from iodine-131 because of its low MPC, and because its long half-life relative to the other iodine isotopes results in its greater concentration in the liquid. The accident is assumed to occur at zero load when the steam generators contain maximum water. With allowance for plate-out retention in water droplets, one-tenth of the contained iodine is assumed released from the plant. The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot X/Q \cdot DCF$$

where: C = secondary coolant activity, 1.0 uCi/cc

V = water volume in one steam generator, 3510 ft^3 = 99 m^3

B(t) = breathing rate, 3.47×10^{-4} m^3/sec

X/Q = 2.9×10^{-4} sec/m^3

DCF = 1.48×10^6 rem/Ci iodine-131 inhaled

The resultant dose is less than 1.5 rem.

Turbine Overspeed Protection

Turbine overspeed protection is provided to limit the possibility of turbine missiles. Overspeed protection is provided by three independent systems based on diverse operating principles. The three systems are the electro-hydraulic (E-H) system, the mechanical trip system, and the Redundant Overspeed Trip System (ROST). The E-H and mechanical systems are single channel and operate on a one-out-of-one to trip logic; the ROST system is a three channel system, requiring two out of three channels to trip.

References:

FSAR Section 10

FSAR Section 14.1

Each relay in the undervoltage protection channels will fail safe and is alarmed to alert the operator to the failure.

A blackout signal which occurs during the sequence loading following a safety injection signal will result in a reinitiation of the sequence loading logic at time step 0 as long as the Safety Injection signal has not been re-set. The Kewaunee Emergency Procedures warn the operators that a Blackout Signal occurring after reset of Safety Injection will not actuate the sequence loading and instructs to re-initiate Safety Injection if needed.

Instrument Operating Conditions

During plant operations, the complete protective instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection Systems, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service 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 out of service.

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 on process control equipment 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. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in another channel.

The operability of the instrumentation noted in Table TS 3.5-6 assures that sufficient information is available on these selected plant parameters to aid the operator in identification of an accident and assessment of plant conditions during and following an accident. In the event the instrumentation noted in Table TS 3.5-6 is not operable, the operator is given instruction on compensatory actions.

References:

(1) FSAR Section 7.5
(2) FSAR Section 14.3
(3) FSAR Section 14.2.5
(4) Deleted

CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objective

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the hot shutdown margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron, or part length rod position.

b. Power Distribution Limits

1. At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

A. $F_Q^N(Z)$ Limits:

(i) Westinghouse Electric Corporation Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \le (2.14)/P \times K(Z) \text{ for } P > .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \le (4.28) \times K(Z) \text{ for } P \le .5$$

(ii) Exxon Nuclear Company Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \le (2.28)/P \times K(Z) \text{ for } P > .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \le (4.56) \times K(Z) \text{ for } P \le .5$$

Proposed Amendment No. 76
08/01/86

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
9	Safeguards Bus Undervoltage (4)	Loss of Power	85.0% ± 2% nominal bus voltage ≤ 2.5 second time delay
10	Safeguards Bus Second Level (5) Undervoltage	Degraded Grid Voltage	92.5% ± 2% of nominal bus voltage ≤ 5 minutes time delay

(1) Initiates containment isolation, feedwater line isolation shield building ventilation, auxiliary building special vent, and starting of all containment fans. In addition, the signal overrides any bypass on the accumulator valves.

(2) Confirm main steam isolation valves closure within 5 seconds when tested.
d/p = differential pressure

(3) The setting limits for max radiation levels are derived from the technical specification 7.4.1, Table E of the OOCM, and section 6.5 of the USAR.

(4) This undervoltage protection channel ensures ESF equipment will perform as assumed in the FSAR.

(5) This undervoltage protection channel protects ESF equipment from long term low voltage operation.

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP
(Page 1 of 3)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	Manual	2	1	1	-		Maintain hot shutdown
2	Nuclear Flux Power Range(3)						
	low setting	4(2)	2	3	1	P-10	Maintain hot shutdown
	high setting	4(2)	2	3	1		
	Positive Rate	4(2)	2	3	1		
	Negative Rate	4(2)	2	3	1		
3	Nuclear Flux Intermediate Range	2	1	1	-	P-10	Maintain hot shutdown - (1)
4	Nuclear Flux Source Range	2	1	1	-	P-6	Maintain hot shutdown - (1)
5	Overtemperature ΔT	4(2)	2	3	1		Maintain hot shutdown
6	Overpower ΔT	4(2)	2	3	1		Maintain hot shutdown
7	Low Pressurizer Pressure	4(2)	2	3	1	P-7	Maintain hot shutdown

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP
(Page 2 of 3)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
8	High Pressurizer Pressure	3	2	2	-		Maintain hot shutdown
9	Pressurizer High Water Level	3	2	2	-	P-7	Maintain hot shutdown
10	Low Flow In one Loop	3/loop	2/loop (any loop)	2	-	P-8	Maintain hot shutdown
	Low Flow Both Loops	3/loop	2/loop (any loop)	2	-	P-7	Maintain Hot shutdown
11	Deleted						
12	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	-		Maintain hot shutdown
13	Undervoltage 4-KV Bus	2/bus	1/bus (both buses)	1/bus	-	P-7	Maintain hot shutdown
14	Underfrequency 4-KV Bus(4)	2/bus	1/bus (both buses)	1/bus	-		Maintain hot shutdown
15	Deleted						
16	Steam Flow/Feedwater Flow Mismatch	2	1	1	-		Maintain hot shutdown

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

NOTES

P-6: 1 of 2 Intermediate Range Nuclear Instrument Channels greater than 10^{-10} amps.

P-7: 3 of 4 Power Range Nuclear Instrument channels less than 10% power AND 2 of 2 Turbine Impulse Pressure channels less than 10% power.

P-8: 3 of 4 Power Range Nuclear Instrument Channels less than 10% power.

P-10: 2 of 4 Power Range Nuclear Instrument Channels greater than 10% power.

Note 1: When a block condition exists, maintain normal operation.

Note 2: when one channel is out of service, a bypass may be used to allow testing other channels; a channel shall not be bypassed longer than 4 hours.

Note 3: One additional channel may be taken out of service for zero power physics testing.

Note 4: Underfrequency on the 4 kV Buses trips the Reactor Coolant Pump breakers, which in turn trips the reactor when power is above P-7.

TABLE TS 3.5-3 (Page 1 of 2)

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	SAFETY INJECTION						
	a. Manual	2	1	1	-		Hot Shutdown***
	b. High Containment Pressure	3	2	2	-		Hot Shutdown***
	c. Low Steam Pressure/Line	3	2	2	-	Primary pressure < 2000 psig	Hot Shutdown***
	d. Pressurizer Low Pressure	3	2	2	-	Primary pressure < 2000 psig	Hot Shutdown***
2	SELECTED BORIC ACID STORAGE TANK LEVEL	2 sets of 2	1 of 2 in each set	2 per set	1/set		One channel may be inoperable for 72 hours otherwise maintain cold shutdown
3	CONTAINMENT SPRAY						
	a. Manual	2	2	2	**		Hot Shutdown***
	b. Hi-Hi Containment Pressure (Containment Spray)	3 sets of 2	1 of 2 in each set	1 per set	1/set		Hot Shutdown***

(Deleted)

TABLE TS 3.5-6

INSTRUMENTATION OPERATING CONDITIONS FOR INOICATION

NO.	FUNCTIONAL UNIT	1 REQUIRED TOTAL NO. OF CHANNELS*	2 MINIMUM CHANNELS OPERABLE**
1	Auxiliary Feedwater Flow to Steam Generators (Narrow Range Level Indication already required operable by Tech Spec Table TS 3.5-2 Item 12)	1/steam gen	1/steam gen
2	Reactor Coolant System Subcooling Margin	2	1
3	Pressurizer Power Operated Relief Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve	1/valve
4	Pressurizer Power Operated Relief Block Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve)	2/valve	1/valve
5	Pressurizer Safety Valve Position (One Channel Temperature, and one Acoustic Sensor per valve)	2/valve	1/valve
6	Containment Water Level (wide range)	2	1
7	Containment Hydrogen Monitor	2	1
8	Containment Pressure Monitor (wide range)	2	1

*With the number of Operable monitoring instrumentation channels less than the Required Total Number of Channels shown, either restore the inoperable channels to Operable status within fourteen days, or be in at least Hot Shutdown within the next 12 hours.

**With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirements, either restore the minimum number of channels to Operable status within 72 hours or be in at least Hot Shutdown within the next 12 hours.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager has overall on-site responsibility for plant operation. In the absence of the Plant Manager, the succession to this responsibility shall be in the following order:

a. Assistant Manager-Plant Operations
b. Assistant Manager-Plant Maintenance
c. Superintendent-Plant Operations
d. Assistant Manager-Plant Technical and Services
e. Shift Supervisor

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for plant management and technical support shall be as shown on Figure TS 6.2-1.

FACILITY STAFF

6.2.2 The plant organization shall be as shown on Figure TS 6.2-2 and:

a. Each on-duty shift complement shall consist of at least:
   (1) One Shift Supervisor (SRO)
   (2) Two licensed Reactor Operators
   (3) One Auxiliary Operator
   (4) One Equipment Operator
   (5) One Radiation Technologist

b. While above cold shutdown, the on-duty shift complement shall consist of the personnel required by 6.2.2a. above and an additional SRO.

c. In the event that one of the shift members becomes incapacitated due to illness or injury or the Radiation Technologist has to accompany an injured person to the hospital, reactor operations may continue with the reduced complement until a replacement arrives. In all but severe weather conditions, a replacement is required within two hours.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of operating, maintenance and other procedures including emergency operating procedures which affect nuclear safety as determined by the plant manager. Changes to those procedures are made in accordance with the provisions of TS 6.8.1.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Review of all proposed changes to the Security Plan and Emergency Plan and their respective implementing procedures.

f. Review all reports covering the investigation of all violations of the Technical Specifications and the recommendations to prevent recurrence.

g. Review plant operations to detect potential safety hazards.

h. Performance of special reviews and investigations and prepare reports thereon as requested by the Chairman of the Nuclear Safety Review and Audit Committee.

f. Reports covering significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. Reports covering all Reportable Events.

h. Reports covering any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the PORC.

AUDITS

6.5.3.8 Audits of plant activities shall be performed under the cognizance of the NSRAC; these audits shall include:

a. Conformance of plant operation to the provisions contained within the Technical Specifications and applicable license conditions at least annually.

b. Performance, training, and qualifications of the entire plant staff at least annually.

c. Results of all actions taken to correct deficiencies occurring in plant equipment, structures, systems, or method of operation that affect nuclear safety at least semiannually.

d. Performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once every two years.

e. Deleted.

f. The Plant Fire Protection Program, implementing procedures and the independent fire protection and loss prevention program at least once every 24 months.

4. Pre-planned procedures and back-up instrumentation to be used if one or more monitoring instruments become inoperable.
5. Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

6.15 SECONDARY WATER CHEMISTRY

The licensee shall implement a secondary water chemistry monitoring program. The intent of this program will be to control corrosion thereby inhibiting steam generator tube degradation. The secondary water chemistry program shall act as a guide for the chemistry group in their routine as well as non-routine activities.

6.16 RADIOLOGICAL EFFLUENTS

Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. Process Control Program implementation.
b. Offsite Dose Calculation Manual implementation.
c. Quality Assurance Program for effluent and environmental monitoring.

6.17 PROCESS CONTROL PROGRAM (PCP)

6.17.1 The PCP shall be approved by the Commission prior to implementation.
6.17.2 Licensee initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   1. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information;

   2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing cri-

SPECIFICATIONS

7.4.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the SITE BOUNDARY shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3.c, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENTS

8.4.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM once per 31 days.

TABLE 8.4 (CONTINUED)

b The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.3.b.

c The ratio of the sample flow rate to the sampled stream flow rate shall be known (based on sampler and ventilation system flow measuring devices or periodic flow estimates) for the time period covered by each dose or dose rate calculation made in accordance with Specifications 7.4.1, 7.4.2 and 7.4.3.