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March 26, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Technical Specifications
NRC Docket Nos. 50-454 and 50-455

References (a): January 20, 1983 letter from T. R. Tramm to H. R. Denton.

(b): December 16, 1983 memorandum from Cecil O. Thomas.

Dear Mr. Denton:

This is to provide comments and suggestions regarding the proof and review version of the Byron 1 Technical Specifications that was distributed in reference (b). NRC review of specific changes proposed here is necessary before the Technical Specifications can be finalized.

Attachments A through O to this letter contain marked-up pages of various sections of the Technical Specifications. A summary explanation of the changes is provided for each attachment. Justifications are provided where appropriate.

We understand that the NRC will review each of these proposed changes and inform Commonwealth Edison of their acceptability. A few additional changes will be submitted by April 2, 1984. After that date, we will be available to meet with the NRC in Bethesda to review these proposed changes and any others which have arisen in Staff review of the proof and review version.

To accomodate the possibility of a favorable decision on our appeal of the ASLB Initial Decision, it is apparent that the Byron 1 Technical Specifications must be finalized by the end of April. We will make every effort to support that schedule.

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Please direct any questions you may have regarding this matter to this office.

One signed original and fifteen copies of this letter and the attachments are provided for NRC review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

lm

cc: Byron Resident Inspector

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Attachment A
(Definition 1.0)

1) Definition 1.5 (Pg 1-1) Channel Calibration

"With" has been changed to "within" for clarification.

2) Definition 1.16 (Pg 1-3) Master Relay Test

The phrase "as a minimum" has been added, after the word "include" for clarification.

3) Definition 1.20 (Pg 1-4) Operational Mode-Mode.

"Reactor" has been deleted for consistency with Technical Specification Table 1.2.

4) Definition 1.25 (Pg 1-5) Quadrant Power Tilt Ratio.

The phrase "shall be used" is changed to "are used" to reflect a definition instead of an action statement. This change is more in line with a definition statement.

5) Definition 1.29 (pg. 1-5) Shutdown Margin

The "full-length" of "full-length rod cluster assemblies" has been deleted because there is no need for rod length designations at Byron.

6) Definition 1.35 (Pg 1-6) Staggered Test Basis

"n" has been defined more clearly by modifying the statement to read "... a given number, n, of . . ."

7) Definition 1.38 (Pg 1-6) Unrestricted Area

"Residual" was incorrectly typed and replaced by the word "residential."

(0423M)

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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within ~~with~~ the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

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$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include as a minimum a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors and persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average ~~reactor~~ coolant temperature specified in Table 1.2. X

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive wastes.

PURGE - PURGING

1.24 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors ~~shall be used~~ are used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full-length~~ rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.32 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for ~~n~~ a given number, n, of systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for ~~residual~~ residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

Attachment B
(Section 2.0)

1) Section 2.1.2 (Pg. 2-1) Reactor Coolant System Pressure

The suggested change to "this" from "its" is necessary for clarity.

2) Section 2.2.1 (Pg. 2-4) Reactor Trip System Instrumentation Setpoints

"S" has been removed from the word Interlocks and "with" has been changed to "within" for clarification.

3) Section 2.2.1 (Pg. 2-4) Reactor Trip System Instrumentation Setpoints

In order to avoid a confusion, "R" and "S" have been changed to "RE" and "SE". S is a abbreviation for Laplace Transform and Shiftly (8 hr). R is an abbreviation for Refueling.

4) Table 2.2-1 (Pg. 2-6, 2-8, 9, 10, 11) Reactor Trip System Instrumentation Trip Setpoints

For Functional Unit "Steam Generator Water Level Low-Low" (Number 13), Z = 18.28 and the Allowable Value is $\geq$ 39.1% instead of 36%; the "Trip Setpoint for Undervoltage - Reactor Coolant Pumps" (number 14) is $\geq$ 4920 volts - each bus; the Trip Setpoint for "Underfrequency - Reactor Coolant Pumps" (number 15) is $\geq$ 57.0 Hz and the Allowable Value is 52.6 Hz; The value K_1 = 1.48 on page 2-8; 3.3% replaced 2.5% in "Note 2" on page 2-9; K_4 = 1.072 on page 2-10; and 2.6% replaced 2% in "Note 4" on page 2-11.

The suggested changes are made based on comments from Westinghouse.

5) Basis Section 2.2.1 (Pg. B 2-3)

In order to avoid a confusion, "R" and "S" have been changed to "RE" and "SE". S is a abbreviation for Laplace Transform and Shiftly (8 hr). R is an abbreviation for Refueling.

6) Basis Section 2.2.1 (Pg. B 2-4) Reactor Trip System Instrumentation Setpoints

"Bypass" has been added to make the statement more specific to Byron Station. The word "addition" replaces "insertion" to make the statement specific to Byron Station.

7) Basis Section 2.2.1 (Pg. B 2-8) Turbine Trip

For clarification, "pressure" has been added to the statement.

8) Basis Section 2.2.1 (Pg. B 2-9) Reactor Trip System Interlocks

The suggested change clearly states two independent trips for RCP undervoltage and underfrequency. The statement becomes ". . . undervoltage, reactor coolant pump bus underfrequency . . .".

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.5.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

this X

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within ~~its~~ limit within 1 hour, and comply with the requirements of Specification 6.5.1.

MODES 3, 4 and 5:

this

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within ~~its~~ limit within 5 minutes, and comply with the requirements of Specification 6.5.1. X

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock~~s~~ Setpoints shall be set consistent ~~with~~ within the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

1. Determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel and adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1, or

2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R\cancel{E} + S\cancel{E} \leq TA$$

Where:

Z = The value for Column Z of Table 2.2-1 for the affected channel,

R~~E~~ = The "as measured" value (in percent span) of rack error for the affected channel,

S~~E~~ = Either the "as measured" value (in percent span) of the sensor error, or the value for Column S~~E~~ (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value for Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop design flow*	≥89.2% of loop design flow*
13. Steam Generator Water Level Low-Low	27.1	18.28 ~~27.18~~	1.5	≥40.8% of narrow range instrument span	39.1% ≥~~36%~~ of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	3.3	0	0	4920 ≥~~4890~~ volts - each bus	≥4768 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	13.3	0	57.0 ≥~~57.5~~ Hz	52.6 ≥~~57.6~~ Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥45 psig	≥43 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.

*Loop design flow = 95,700 gpm

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT = Measured ΔT by RTD Manifold Instrumentation,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s,

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT,

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s,

ΔT_0 = Indicated ΔT at RATED THERMAL POWER,

X K_1 = ~~1.072~~ 1.48,

K_2 = 0.0265/°F,

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 33$ s, $\tau_5 = 4$ s,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s,
T'	$\leq$	587.7°F (Nominal T_{avg} at RATED THERMAL POWER),
K_3	=	0.00134,
P	=	Pressurizer pressure, psig,
P'	=	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -42% and -6%, $f_1(\Delta I)$ = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -42%, the ΔT Trip Setpoint shall be automatically reduced by 2.86% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -6%, the ΔT Trip Setpoint shall be automatically reduced by 1.86% of its value at RATED THERMAL POWER.

X NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.5%~~ 3.3 ΔT ~~instrument~~ span.

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

X K_4 = ~~1.083~~ 1.072,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00170/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 587.7°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~(2)~~% ΔT ~~instrument~~ span.
2.6

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2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R\cancel{E} + S\cancel{E} \leq TA$, the interactive affects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R~~E~~ or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S~~E~~ or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

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REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip & bypass breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore, providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity ~~insertion~~ that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

addition

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

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Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber Pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7. X

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position trips are anticipatory trips which provide core protection against DNB. The Open/Close Position trips assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Trip System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent) an automatic Reactor trip will occur if more than one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7 an automatic Reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range Reactor trip (i.e., prevents premature block of Source Range trip), provides a backup block for Source Range Neutron Flux doubling, and de-energizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage, ~~and~~ reactor coolant pump bus underfrequency, Turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip; and automatically blocks the Source Range Reactor trip and de-energizes the Source Range high voltage ~~power~~. On decreasing power, the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

Attachment C
(Section 3/4.1)

1) Section 3.1.1.1 (Pg. 3/4 1-1) Boration Control Shutdown Margin

   Clarification of "equivalent" is suggested in order to make the statement clear. The proposed change states "or equivalent boration rate".

2) Section 4.1.1.1.1 (Pg. 3/4 1-1) Shutdown Margin

   The word "rod" should be "rod(s)" in section 4.1.1.1.1.a to agree with the format of the paragraph.

   Deletion of "below" is requested for clarification in Section 4.1.1.1.1.d..

3) Section 4.1.1.1.2 (Pg. 3/4 1-2) Shutdown Margin

   Deletion of "above" is requested for clarification.

4) Section 3.1.1.2 (Pg. 3/4 1-3) Shutdown Margin

   Clarification of "equivalent" is suggested in order to make the statement clear. The proposed change states "or equivalent boration rate".

5) Section 4.1.1.2.a 9 (Pg. 3/4 1-3) Shutdown Margin

   The word "rod" should be "rod(s)" to agree with the format of the paragraph.

6) Section 3.1.1.3.a & b (Pg. 3/4 1-4) Moderator Temperature Coefficient

   Deletion of "above" is requested for clarity in Action Statement 3.1.1.3.a & b.

7) Section 4.1.1.3.a (Pg. 3/4 1-5) Moderator Temperature Coefficient

   Deletion of "above" is request for clarity.

8) Section 3/4.1.3 (Pg. 3/4 1-14, 15) Movable Control Assemblies

   "Full length" has been deleted in Sections 3.1.3.1, 3.1.3.1.a, 3.1.3.1.b, 3.1.3.1.c. 4.1.3.1.1 and 4.1.3.1.2. Full length rods are not applicable to Byron Station.

9) Section 4.1.3.1.1 (Pg. 3/4 1-15) Movable Control Assemblies

   The suggested change to "alarm" from "monitor" is necessary because the instrument is an alarm, not a monitor.

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10) Table 3.1-1 (Pg. 3/4 1-16) Accident Analyses

"Full length" has been deleted. There is no need for rod length designations at Byron.

11) Section 4.1.3.2 (Pg. 3/4 1-17) Position Indication Systems - Operating

The suggested change to "alarm" from "monitor" is necessary because the instrument is an alarm, not a monitor.

12) Section 3.1.3.4 (Pg. 3/4 1-19) Rod Drop Time

"Full length" has been deleted for Section 3.1.3.4 and Action Statement 3.1.3.4.a.. There is no need for rod length designations at Byron.

13) Section 3.1.3.6.b (Pg. 3/4 1-21) Control Rod Insertion Limits

"3.1-1 and 3.1-2" have been added to make the statement more clear.

14) Section 4.1.3.6 (Pg. 3/4 1-21) Control Rod Insertion Limits

The suggested change to "alarm" from monitor" is necessary because the instrument is an alarm, not a monitor.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Δk/k for four loop operation.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent ^ boration rate until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod(s) is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); X

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. ~~below~~, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exception 3.10.1.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e. ~~above~~. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. 2

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SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent boration rate until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod(s) is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and X

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition, or

b. Less negative than -4.1×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, and of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. ~~above~~, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^{\circ}F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. In lieu of any other report required by Specification 6.7.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.7.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. ~~above~~, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

#See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., ~~above,~~ prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading, and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All ~~full-length~~ shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more ~~full-length~~ rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one ~~full-length~~ rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one ~~full-length~~ rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 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;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions 3.10.2 and 3.10.3.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) 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; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each ~~full-length~~ rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation ~~monitor~~ is inoperable, then verify the group positions at least once per 4 hours. alarm

4.1.3.1.2 Each ~~full-length~~ rod not fully inserted in the core shall be determined OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

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TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ~~FULL-LENGTH~~ ROD

Rod Cluster Control Assembly Insertion Characteristics.

Rod Cluster Control Assembly Misalignment.

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System.

Single Rod Cluster Control Assembly Withdrawal at Full Power.

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident).

Major Secondary Coolant System Pipe Rupture.

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection).

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating 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

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one bank demand position indicator inoperable either:

   1. Verify that all digital 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

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined OPERABLE by verifying that the Bank Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation ~~monitor~~ alarm is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual ~~full length~~ shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 550°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod 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 proceeding to MODE 1 or 2.

b. With the rod drop time within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. 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, and

c. At least once per 18 months.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the ~~above~~ figures~~, or~~ 3.1-1 and 3.1-2

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit ~~Monitor~~ is inoperable, then verify the individual rod positions at least once per 4 hours.

Alarm

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

Attachment D
(Section 3/4.2)

1) Section 3/4.2.1 (Pg. 3/4 2-1) Axial Flux Difference

   The "#" character has been moved from Action Statement 3.2.1.b to the LCO so the operator will be aware of the note prior to entering an Action Statement.

2) Section 4.2.2.2 (Pg. 3/4 2-6) Power Distribution Limits

   4.2.2.2.b statement is ambiguous until 4.2.2.2.c is read. The proposed change adds "calculating F_{xy} by increasing . . ." at the beginning of the surveillance 4.2.2.2.b. The words "above" and "below" have been deleted for clarification of Technical Specification format.

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3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU, and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

X

The indicated AFD may deviate outside the above required target band at greater than or equal to 50%# but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

   1. Restore the indicated AFD to within the above required target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux# - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours. X

*See Special Test Exception 3.10.2.

#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

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4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER;

Calculating F^c_{xy} by

b. ~~I~~ncreasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties;

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2b., ~~above~~, to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specifications 4.2.2.2e. and f., ~~below~~, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} [1+0.2(1-P)]$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^{L}:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

Attachment E
(Section 3/4.3)

1) Table 3.3-1 (Pg. 3/4 .-3) Reactor Trip System Instrumentation

The Action for Turbine Trip - Emergency Tripheader EHC, Turbine Stop valve closure (Functional Unit 16b). has been changed from #14 to #11 to reflect correct Action.

2) Table 3.3-2, 4.3-1 & 3.3-3 (Pg. 3/4 3-7~14) Reactor Trip System Instrumentation Surveillance.

The statement "(Above P-7)" has been added to Functional Unit numbers 9, 11, 14, 15, 16, 18; 9, 11; and 1.d for consistency in Technical Specification format.

"Turbine Stop Valve" has been changed to "Turbine Throttle Valve". Turbine Throttle Valve is the station approved name.

3) Table 3.3-3 (Pg. 3/4 3-16) Engineered Safety Features Actuation System Instrumentation.

The Action for Functional Unit "Steamline Isolation, Manual Initiation - Individual" (4.a.1) has been changed from 24 to 23 to reflect correct action.

4) Table 3.3-4 (Pg. 3/4 3-24~27) Engineered Safety Features Actuation System Instrumentation Trip Setpoints.

Functional "Steamline Pressure-Low" Unit (4.d) added "(Above P-11)", "Functional Unit Steamline Pressure - Negative Rate - High" (4.e) added "(Below P-11)", "Functional unit ESF Bus Undervoltage" (8.a) added "(Electro-mechanical Relaying)" and "Functional Unit Grid Degraded Voltage" (8.b) added "(Solid State Relaying)" to the respective description for consistency in Technical Specification format.

5) Table 4.3-2 (Pg. 3/4-33~36) Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

For consistency in Technical Specification format; Functional Unit "Steamline Pressure - Low" (1.e) and "Steamline Pressure - Low" (4.d) added "(Above P-11) "(Below P-11)" to "Steamline Pressure - Negative Rate-High" (4.e); "(P-14)" to "Steam Generator Water Level - High-High" (5.b); "(Electro-mechanical Relaying)" to "ESF Bus Undervoltage" (8.a) and "(Solid State Relaying)" to "Grid Degraded Voltage" (8.b).

6) Table 3.3-11 (Pg 3/4 3-57) Fire Detection Instrument

Zones 41 through 56 were added to number 4 "Upper Cable Spreading Room" and "Lower Cable Spreading Room". (see attachment C.a). These additional monitors are also used to meet Technical Specification requirements.

Attachment E (Continued)
(Section 3/4.3)

7) Table 3.3-11 (Pg. 3/4 3-58) Fire Detection Instrument

There are two ultra violet (flame) detectors in zones 71 and 72 in the Diesel Generator Room. "Zone 71" and "Zone 72" with "Detection" and Flame "2" were added to number 7, "Diesel Generator Room".

8) Table 3.3-12 (Pg. 3/4 3-61) Radioactive Liquid Effluent Monitoring Instrumentation

To indicate and describe Radioactivity Monitors providing alarm but not providing Automatic Termination of Release, the subscripts read a. Essential Service Water RCFC 1A and 1C Outlet (1RE-PR002), b. Essential Service Water RCFC 1B and 1D Outlet (1RE-PR003) and c. Station Blowdown Line (0RE-PR010).

9) Table 4.3-8 (Pg. 3/4 3-63) Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirement

To indicate and describe Radioactivity Monitors providing alarm but not providing Automatic Termination of Release, the subscripts read a. Essential Service Water RCFC 1A and 1C Outlet (1RE-PR002), b. Essential Service Water RCFC 1B and 1D Outlet (1RE-PR003) and c. Station Blowdown Line (0RE-PR010).

10) Table 4.3-8 (Pg 3/4 3-64) Table Notations

The typographical error "DITIGAL" is corrected to "DIGITAL" in Table Notation 1.

11) Section 3/4.3.4 (Pg. 3/4 3-74) Turbine Overspeed Protection

Exchanged the Surveillance Requirement items 4.3.4.2.b and 4.3.4.2.c to clarify which valves are being discussed.

Turbine Stop valves at Byron Station are referred to as Turbine Throttle valves.

Changed the Surveillance Requirement 4.3.4.2.a to a frequency of 31 days and incorporate the former item 4.3.4.2.c of "direct observation of the movement" into the statement. The basis for this change is the Westinghouse Operation and Maintenance Memo 041 Recommended Testing Frequency For Steam Admission Valves on BB296 Nuclear Turbines With Steam Chests.

Added the following basis for supporting frequency, methods of verification and reasons for changes to the surveillance requirements.

A copy of the surveillance requirements is included showing all change which were discussed in this letter.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1, 2	6#
11. Pressurizer Water Level-High (Above P-7)	3	2	2	1	7#
12. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
13. Steam Generator Water Level-Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6#
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6#
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6#
16. Turbine Trip -Emergency Tripheader EHC (Above P-7)					
a. Low Auto Stop Oil Pressure	3	2	2	1	7#
b. Turbine Stop Valve Closure	4	4	1	1	~~14#~~ ~~6#~~ 11#

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TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$ 6 seconds*
8. Overpower ΔT	N.A.
9. Pressurizer Pressure-Low (Above P-7)	$\leq$ 2 seconds
10. Pressurizer Pressure-High	$\leq$ 2 seconds
11. Pressurizer Water Level-High (Above P-7)	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow - Low	
a. Single Loop (Above P-8)	≤ 1 second
b. Two Loops (Above P-7 and below P-8)	≤ 1 second
13. Steam Generator Water Level-Low-Low	≤ 2 seconds
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	≤ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	≤ 0.6 second
16. Turbine Trip (Above P-7)	
a. Low Auto Stop Oil Pressure	N.A.
b. Turbine ~~Stop~~ Throttle Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.
19. Reactor Trip System Interlocks	N.A.
20. Reactor Trip Breakers	N.A.
21. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5, 12)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (Above P-7)	S	R	M	N.A.	N.A.	1
12. Reactor Coolant Flow-Low	S	R	M	N.A.	N.A.	1

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	1, 2
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	M	N.A.	1
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	M	N.A.	1
16. Turbine Trip (Above P-7)						
a. Low Auto Stop Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine ~~Stop~~ Valve Closure Throttle	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M (8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M (8)	N.A.	N.A.	1

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure-Low (Above P-7)	4	2	3	1, 2, 3#	18*
e. Steam Line Pressure-Low (above P-11)	3/stm. gen.	2/stm. gen. any steam line	3/stm. gen.	1, 2, 3#	15*
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair	2 pair	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-3	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/stm line	1/stm line	1/operating steam line	1, 2, 3	~~24~~ 23
2) System	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure-High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure-Low (above P-11)	3/stm. gen.	2/stm. gen. any steam line	3/stm. gen.	1, 2, 3#	15*
e. Steam Line Pressure - Negative Rate-High (below P-11)	3/stm. gen.	2/stm. gen. any steam line	2/stm. gen.	3##, 4	15*
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2,	21
b. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	15*

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-2	5.0	0.71	1.5	$\leq$ 10.0 psig	$\leq$ 12.0 psig
d. Steam Line Pressure-Low (Above P-11)	14.2	10.71	1.5	$\geq$ 640 psig	$\geq$ 610 psig*
e. Steam Line Pressure-Negative Rate-High (Below P-11)	8.0	0.5	0	$\leq$ -100 psi/s	$\leq$ -110.0 psi/s**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)	5.0	2.18	1.5	$\leq$ 82% of narrow range instrument span	$\leq$ 83% of narrow range instrument span

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. ESF Bus Undervoltage (Electromechanical Relaying)	N.A.	N.A.	N.A.	2870 volts	2870 volts ± 143 1.8 ± 0.1s time delay
b. Grid Degraded Voltage (Solid State Relaying)	N.A.	N.A.	N.A.	3804 volts 9s time delay	3804 volts ± 76 9 ± 0.9s time delay
9. Engineered Safety Feature Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ 1950 psig	$\leq$ 2050 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	550°F	$\geq$ 548°F and $\leq$ 552°F
d. Steam Generator Water Level, P-14 (High-High)	See Item 5.b. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

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TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3,
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3,
e. Steam Line Pressure-Low (Above P-11)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
C. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure - Negative Rate - High (Below P-11)	S	R	M	N.A.	N.A.	N.A.	N.A.	3, 4
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Undervoltage-RCP Bus	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Division 1 ESF Bus Undervoltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A	1, 2, 3
g. Auxiliary Feedwater Pump Suction Pressure-Low	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Opening of Containment Sump Suction Isolation Valves								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With Safety Injection	N.A.	R	N.A.	N.A.	M	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage (Electromechanical Relaying)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage (Solid State Relaying)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	INSTRUMENT TYPE*	TOTAL NUMBER OF INSTRUMENTS Heat	Flame	Smoke
1. Containment ***				
Zone 11 Elev 426	Suppression	1 **		
Zone 12 Elev 426	Suppression	1 **		
Zone 2 Elev 401	Detection			2
Zone 3 Elev 401	Detection			2
Zone 4 Elev 401	Detection			2
Zone 5 Elev 401	Detection			2
Zone 6 Elev 426	Detection			6
Zone 76 Elev 426	Detection			9
Zone 7 Elev 414	Detection			6
Zone 24 Elev 414	Detection			10
2. Control Room				
Zone 68 Elev 451	Detection			3
Zone 69 Elev 451	Detection			8
Zone 75 Elev 451	Detection			10
3. Switchgear Rooms				
Zone 77 Elev 426	Detection			9
Zone 78 Elev 426	Detection			9
4. Upper Cable Spreading Room				
Zone 33 Elev 463	Suppression	Replace with (A)		4
Zone 34 Elev 463	Suppression			8
Zone 35 Elev 463	Suppression			8
Zone 36 Elev 463	Suppression			4
Lower Cable Spreading Room				
Zone 43 Elev 439	Suppression	Replace with (B)		8
Zone 44 Elev 439	Suppression			10
Zone 45 Elev 439	Suppression			8
Zone 46 Elev 439	Suppression			4
5. Remote Shutdown Panel				
Zone 13 Elev 383	Detection			5

(A)

				Smoke
Zone	41	Elev 463	Detection	4
Zone	42	Elev 463	Detection	4
Zone	43	Elev 463	Detection	8
Zone	44	Elev 463	Detection	8
Zone	45	Elev 463	Detection	8
Zone	46	Elev 463	Detection	8
Zone	47	Elev 463	Detection	4
Zone	48	Elev 463	Detection	4

(B)

				Smoke
Zone	49	Elev 434	Detection	8
Zone	50	Elev 434	Detection	8
Zone	51	Elev 434	Detection	10
Zone	52	Elev 434	Detection	10
Zone	53	Elev 434	Detection	8
Zone	54	Elev 434	Detection	8
Zone	55	Elev 434	Detection	4
Zone	56	Elev 434	Detection	4

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	INSTRUMENT TYPE*	TOTAL NUMBER OF INSTRUMENTS Heat	Flame	Smoke
6. Station Battery Room				
Zone 67 Elev 451	Detection			3
7. Diesel Generator Room				
Zone 37 Elev 401	Suppression	4		
Zone 38 Elev 401	Suppression	4		
Zone 71 Elev 401	Detection		1	
Zone 72 Elev 401	Detection		1	
8. Diesel Fuel Storage				
Zone 39 Elev 401	Suppression	1		
Zone 40 Elev 401	Suppression	1		
Zone 27 Elev 383	Suppression	3		~~3~~
Zone 28 Elev 383	Suppression	3		~~3~~
Zone 10 Elev 383	Detection			6
9. Safety Related Pumps				
Zone 41 Elev 383	Suppression	2		
Zone 42 Elev 383	Suppression	1		
Zone 16 Elev 364	Detection			2
Zone 18 Elev 364	Detection			3
Zone 19 Elev 364	Detection			~~3~~ 2
Zone 20 Elev 346	Detection			3
Zone 21 Elev 346	Detection			3
Zone 52 RSH	Suppression	6		
10. Fuel Storage				
Zone 39 Elev 401	Detection			3
Zone 38 Elev 426	Detection		3	

TABLE NOTATIONS

*A single detector in a zone marked "Detection" will alarm in the Main Control Room. A single detector in a zone marked "Suppression" will initiate suppression and alarm in the Main Control Room.

**These are Containment Ventilation temperature switches. Upon receipt of a Hi-Hi temperature, suppression must be manually initiated.

***The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
Liquid Radwaste Effluent Line (0RE-PR001)	1	31
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Essential Service Water ~~Outlet Line~~ RCFC 1A and 1C Outlet (1RE-PR002)	1	32
b. Essential Service Water RCFC 1B and 1D Outlet (1RE-PR003)		
c ~~b~~. Station Blowdown Line (0RE-PR010)	1	32
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (Loop-WX001)	1	33
b. Station Blowdown Line (Loop-CW032)	1	33

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TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
Liquid Radwaste Effluent Line (ORE-PR001)	D	P	R(3)	Q(1)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Essential Service Water ~~Outlet Line~~ RCFC 1A and 1C Outlet (1RE-PR002)	D	M	R(3)	Q(2)
b. Essential Service Water RCFC 1B and 1D Outlet (1RE-PR003)				
c ~~b~~. Station Blowdown Line (ORE-PR010)	D	M	R(3)	Q(2)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line (Loop-WX001)	D(4)	N.A.	R	Q
b. Station Blowdown Line (Loop-CW032)	D(4)	N.A.	R	Q

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The DIGITAL ~~DITIGAL~~ CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ~~stop~~ throttle valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. During turbine operation, at least once per 31 days by direct observation of the movement of the above valves through one complete cycle from the running position.

1) Four high pressure turbine ~~stop~~ throttle valves,
2) Four high pressure turbine governor valves,
3) Six turbine reheat stop valves,
4) Six turbine reheat intercept valves, and

b. Within 7 days prior to entering MODE 3 from MODE 4, by cycling each of the 12 extraction steam nonreturn check valves from the closed position.

c. During turbine operation At least once per 31 days by direct observation, of freedom of movement of each of the 12 extraction steam non return check valve ~~nonreturn check valve~~ weight arms.

d. At least once per 18 months by performance of CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems.

e. At least once per 40 months by disassembling at least one of the valves given in Specification 4.3.4.2a and c above, and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures.

Valves in Specification 4.3.4.2a. (High Pressure Turbine and Reheat Valves)

These valves isolate large quantities of steam with high potential for delivering energy to the rotor system. The turbine design recognizes this potential in providing rapid action, dual shut off capability in each path, remote testing capability, and a flow path that reduces the effects of changes in flow distribution, load reductions and thermal transients during testing. The testing intervals are in accordance with the latest manufacturer's recommendations. (Westinghouse Operation and Maintenance Memo 041, Steam Turbine Division).

Valves in Specification 4.3.4.2b and c. (Extraction Steam Nonreturn Check Valves)

These valves are provided to protect the turbine from reflux of steam remaining in the feedwater heater shells and piping following the pressure reduction caused by the actuation of valves in specification 4.3.4.2a. The quantities of stored steam controlled by these valves are smaller and are divided up into separate heater shells. The feedwater heating system design, including these valves, did not intend routine full stroke testing. The extraction steam check valves are self closing swing disk non return valves which shut under the combined effect of gravity and reverse flow of steam. The weight of the disk is partly balanced by a counterweight and lever on the pivot shaft. A spring cylinder acting on the lever assists the start of the automatic closing, but is not intended to close the valve fully against normal steam flow and pressure. In normal operation the spring assist is held clear by air pressure acting on a piston under the spring. The turbine trip system releases the air pressure to assist the closing.

Manually stroking of the extraction steam non return valves is possible under shutdown conditions by latching the turbine and applying the air pressure to the spring cylinder. It is possible to hear and feel the disk contact the seat solidly. This manual stroking was not provided for in the design but will be done within 7 days prior to entering Mode 3 from Mode 4.

The engineering specification provided for testing and extraction steam non return check valves during operation by equalizing the air pressure across the piston in the spring cylinder, permitting the spring to partially close the disk against the steam flow. The rotation of the shaft accompanying the disk closure can be observed by movement of the weight lever. The amount of movement observed in other stations has depended on the extraction steam conditions and valve size, but has been ample to indicate freedom of movement, and this will be verified during startup testing.

Partial stroking demonstrates that the disk system is free at the beginning of the closing stroke where the steam closing forces are smallest. As the disk enters a reverse steam flow the closing forces build up rapidly with progressive closure.

The design of the feedwater heating system is such that full stroke testing of the extraction steam non return valves during turbine operation involves several penalties without significant additional advantages over partial stroke testing. The motor operated isolating valve must be closed as an individual heater. Heater stages 1, 2, 3 and 4 are arranged in three parallel strings with cascaded drains in each string and heater stages 5, 6 and 7 are similarly arranged in two parallel strings. An entire string is taken out of service, isolated and

bypassed for maintenance. Isolating the extraction steam to a single intermediate heater involves several complications.

The motor operated valves are too large for routine manual operation, do not have bypasses to allow controlled warmup conditions, stroke quickly (about 15 seconds), and are intended for turbine protection against heater flooding. A comparision of the thermal capacity of a heater and the rate of heat transfer to the flowing condensate or feedwater shows that cycling an extraction steam isolating valve would cause rapid cooling and heating transients.

Isolating the steam to a top heater drops the feedwater temperature to the steam generators. Isolating the steam to an intermediate heater causes the next heater to assume the heating load, approximately doubling the steam demand and drain flow, and nearly quadrupling the potential for erosion and vibration in the affected heater and piping. The shell pressure collapses in the isolated heater causing insufficient head to discharge the cascading drains to the next lower heater. Rapid action of the emergency drain control is required to prevent flooding, with the potential for flashing in the drain cooler section from pressure decay.

Isolating a heater degrades the cycle thermal performance, requiring a corresponding drop in electrical output for the same reactor thermal power.

Partial closing of the extraction steam non return check valves with the installed test provisions demonstrates freedom of movement while avoiding transient states. A 31 day interval will be adequate since it is likely that sticking conditions develop during shutdown conditions rather than in operations.

Availability today & tomorrow

Westinghouse

OPERATION & MAINTENANCE MEMO 041

RECOMMENDED TESTING FREQUENCY FOR STEAM ADMISSION VALVES ON BB 296 NUCLEAR TURBINES WITH STEAM CHESTS

NOVEMBER 14, 1983

Approved: RMReber

R. M. Reber, Manager
Technical Development
Power Generation Service Division

Approved: C.W. Meck

C. W. Meck, Product Manager
Commercial Operations
Steam Turbine-Generator Division

OPERATION & MAINTENANCE MEMO 041

1 REASON FOR MEMO

Based on a review of testing frequency and performance data from Westinghouse turbine and component incidents records and a 1982 survey of utilities operating Westinghouse nuclear turbines, Westinghouse concluded that for nuclear units with steam chests there is no significant difference in the valve failure rate between valves tested weekly and those tested monthly. It was further noted that monthly versus weekly valve testing frequency may be beneficial because it reduces the time a plant is operating in a "transient state."

2 OPERATION AND MAINTENANCE INFORMATION

Westinghouse recommends that the throttle, governor, interceptor and reheat stop valves of nuclear turbine-generator units with steam chests be tested monthly.

Attachment F
(Section 3/4.4)

1) Section 3.4.1.2 (Pg. 3/4 4-2) Hot Standby

"And" has been changed to "or" because D Loop is a choice, not a must.

2) Section 3.4.1.3 (Pg. 3/4 4-3) Hot Shutdown

The change has been made because RHR Loop B is a choice, not a must.

3) Section 3.4.2.2 (Pg. 3/4 4-8) Shutdown

"MODE" has been changed to "mode" because the definition does not strictly apply.

4) Section 4.4.5.2 (Pg. 3/4 4-12) Steam Generator Tube Sample Selection and Inspection

The phrase "penetrations (greater than 20%)" has been changed to "degradations (exceeding 20% of wall thickness)", because the tubes may be degraded but not necessarily penetrated.

5) Section 4.4.5.3.c.1 (Pg. 3/4 4-13) inspection Frequencies

"S" has been deleted from "tubes" for clarification.

6) Section 4.4.5.4 (Pg. 3/4 4-14) Acceptance Criteria

"Acceptance Criteria" has been changed to "Definitions" for clarification.

7) Section 4.4.5.5.c (Pg. 3/4 4-15) Reports

6.7.2 is the correct Technical Specification reference.

8) Section 3.4.8 (Pg. 3/4 4-26) Specific Activity

"Example" has been changed to "sample" for clarification.

9) Figure 3.4-3 (Pg. 3/4 4-32) Reactor Coolant System Cooldown Limitations Applicable to 32 EFPY

This has to be changed to incorporate the new closure flange limitations. Attached are the letter and supporting documents from T.R. Tramm to Mr. Harold R. Denton (NRC Docket No.s. 50-454/455 and 50-456/457, written on January 3, 1983). The attachments are on the last page of this package.

10) Section 3.4.9.3 (Pg. 3/4 4-35) Overpressure Protection System

Mode 3 applies to 380°F; therefore, Mode 4 was changed to Mode 3 in the Applicability Statement. Mode 4 was added in the series of applicable modes.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, ~~and~~ or

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side narrow range water level to be greater than or equal to 41% at least once per 12 hours.

4.4.1.2.3 At least one reactor coolant loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All Reactor Coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,**

e. RHR Loop A, and/or

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or RHR loop(s) OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 380°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.2 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling ~~MODE~~. mode

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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SURVEILLANCE REQUIREMENTS (Continued)

1) All nonplugged tubes that previously had detectable wall ~~penetrations (greater than 20%)~~, degradations (exceeding 20% of wall thickness)

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall ~~penetrations~~ degradations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary ~~tubes~~ tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A Condition IV main steam line or feedwater line break.

4.4.5.4 Definitions ~~Acceptance Criteria~~

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., ~~above~~; X

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; ~~and~~ X

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.7.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.7.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification ~~6.7.1~~ 6.7.2 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram of gross radioactivity, perform the sampling and analysis requirements of Item 4a of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

For this ACTION statement, prepare and submit a Special Report to the Commissioner pursuant to Specification 6.7.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

sample

2. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, while limit was exceeded and one ~~example~~ after the radioiodine activity was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentrations,
3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,
4. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and
5. The time duration when the specific activity of the primary coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

Replace

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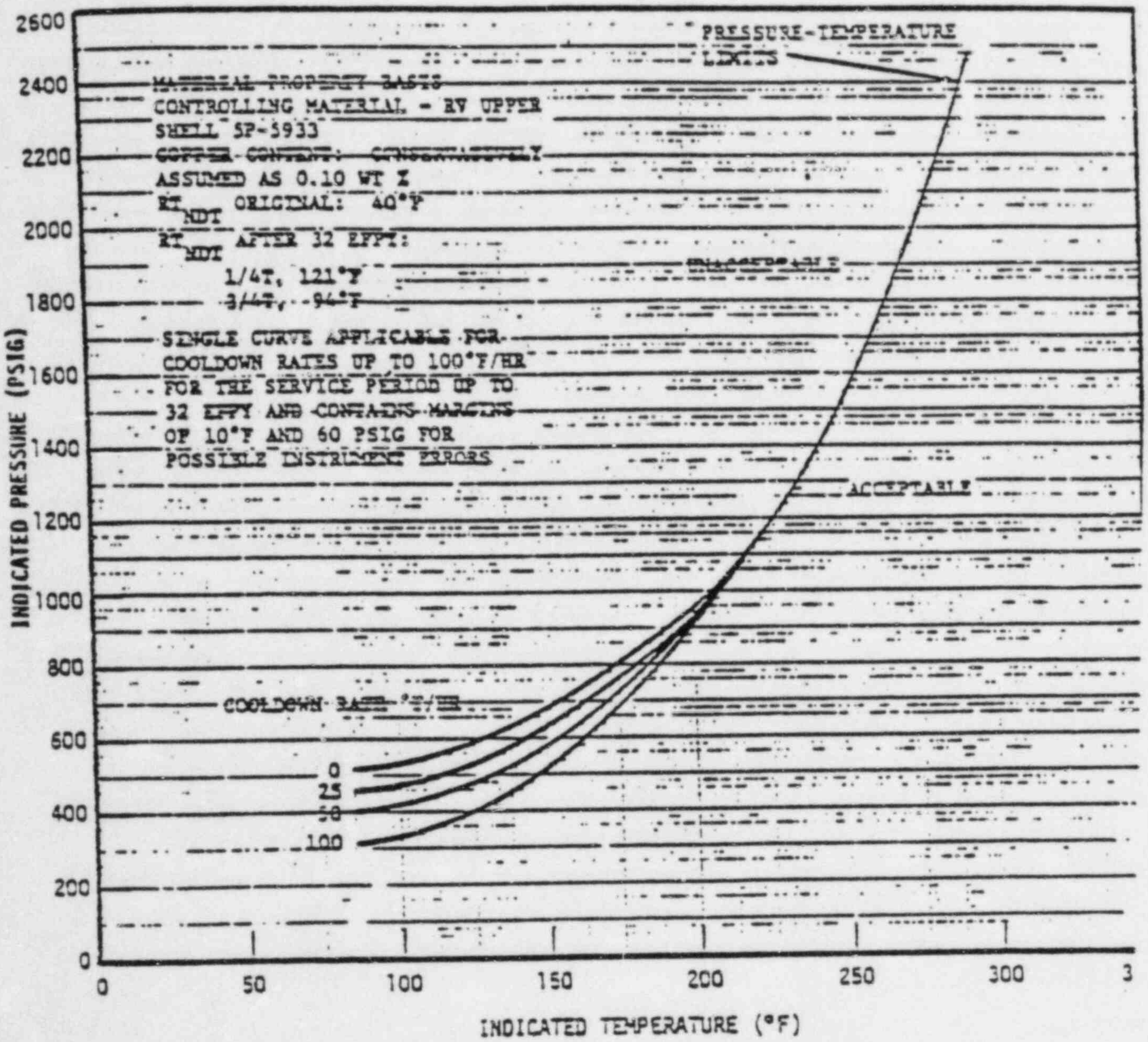
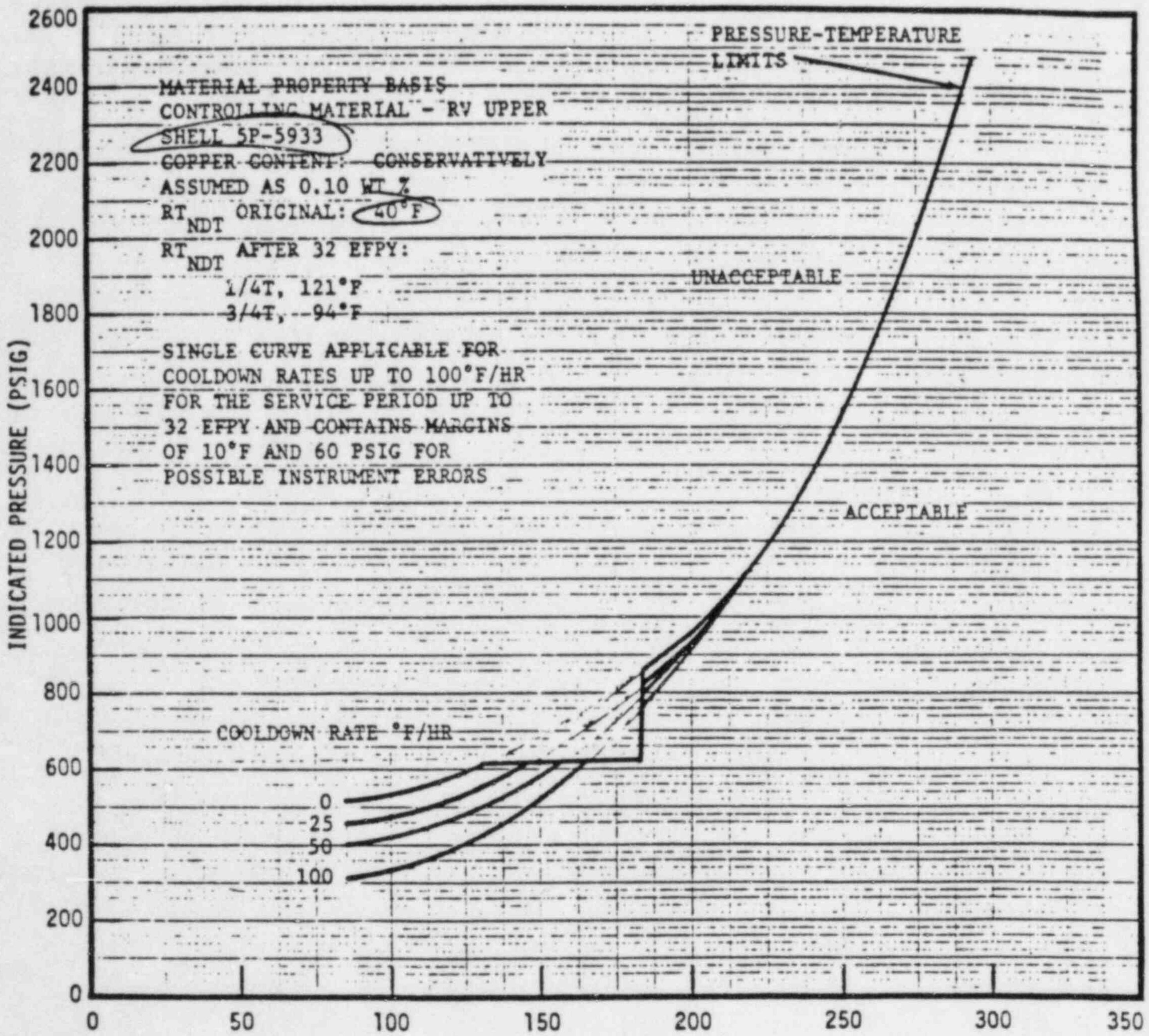


FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 32 EPPY



Unit 1

BYRON STATION

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.4-3a

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE TO 32 EFPY

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with nominal Setpoints which vary with RCS temperature as shown on Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODE ~~4~~ 3 when the temperature of any RCS cold leg is less than or equal to 380°F, MODE 4, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.7.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

January 3, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Braidwood Generating Station Units 1 and 2
Reactor Vessel Temperature Limits
NRC Docket Nos. 50-454/455 and 50-456/457

Dear Mr. Denton:

This is to provide revised temperature limits for the Byron and Braidwood reactor vessels. These revisions were made necessitated by the May 27, 1983 amendment of Appendix G to 10CFR50.

The heatup and cooldown temperature limit curves for all of the Byron/Braidwood reactor vessels have been reviewed against the amended Appendix G requirements. Only the cooldown curves for Byron 1 and Braidwood 2 must be changed to incorporate the new closure flange limitations. Revised copies of those curves (FSAR figures 3.4-3a and 3.4-5b) are enclosed with this letter. Also enclosed is Table I, showing the data used in this review.

These two cooldown curves are to be incorporated into the plant Technical Specifications. They will not be incorporated into the FSAR since it is not our intent to keep current the proposed technical specifications contained in the FSAR.

We understand that the amended Appendix G provides an option for analyzing closure flange region stresses to demonstrate that they are less limiting than the beltline region. If that were done, the special limitations on the enclosed curves would be unnecessary. At some point in the future we expect to pursue that option but for new we will abide by the limits shown in the FSAR, and as shown in the enclosed curves.

Please direct questions regarding this matter to this office.

One (1) signed original and fifteen (15) copies of this letter and the enclosures are provided for NRC review.

Very truly yours,

T. R. Tramm

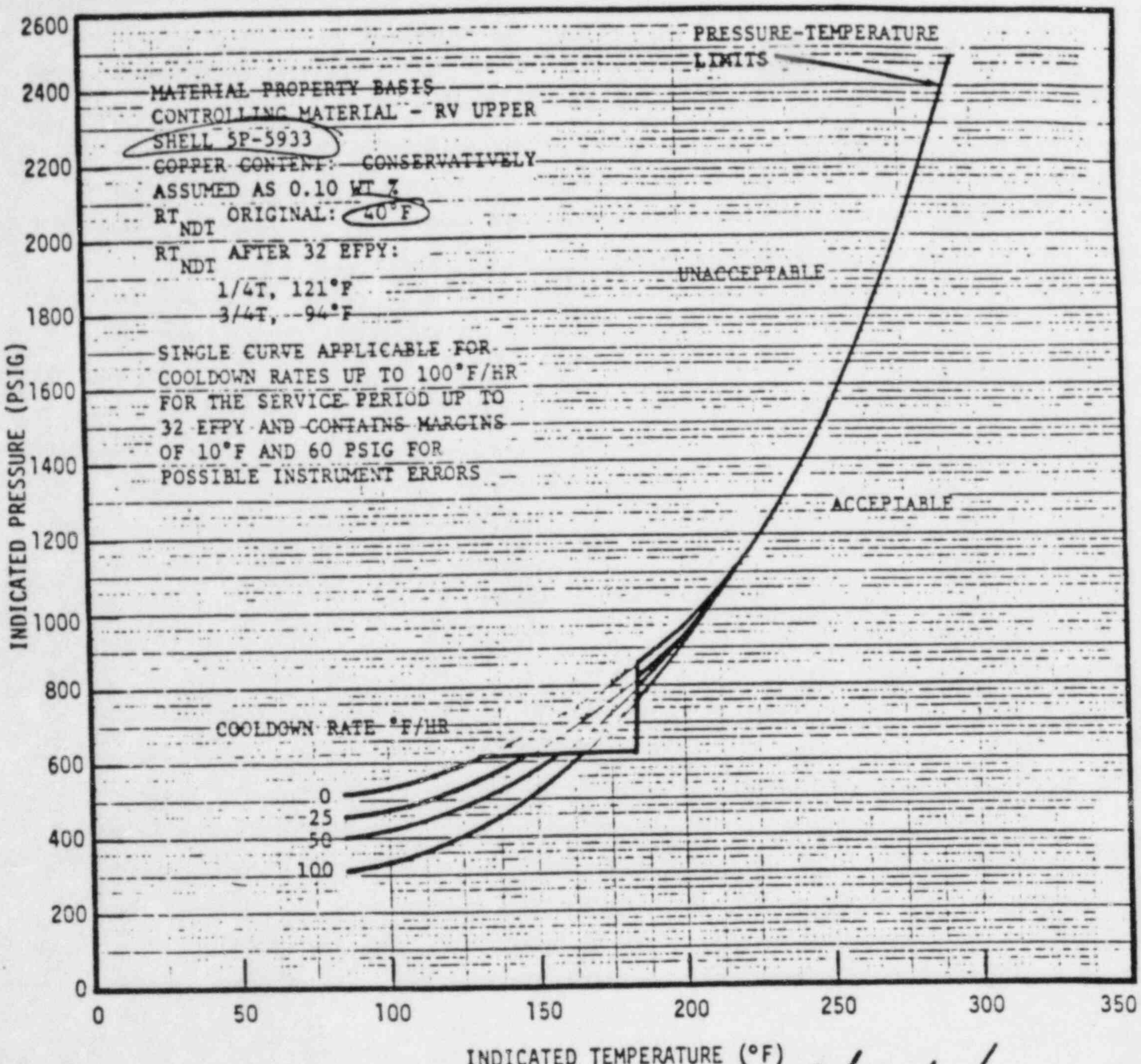
T. R. Tramm
Nuclear Licensing Administrator

Enclosures
7905N

TABLE I: Data used for P-T curves evaluation per 10CFR App. G

Vessel	Closure Flange RT^{1}_{NDT} °F	$(RT_{NDT}+90)$°F	$(RT_{NDT}+120)$°F	Preservice Hydro. Pressure P psig	20%P psig
Byron 1	60	150	180	3107	621.4
Byron 2	0	90	120	3107	621.4
Braidwood 1	-20	70	100	3107	621.4
Braidwood 2	20	110	140	3107	621.4

1. Reference Temperatures, RT_{NDT}, were obtained from B/B FSAR Q251.



Unit 1

BYRON STATION

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.4-3a

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE TO 32 EFPY

Attachment G
(Section 3/4.5)

1) Section 3/4.5.1.d (Pg. 3/4 5-1) Accumulators

A nitrogen cover-pressure of between 600 and 650 psig has been changed to 617 and 662 psig. The correct pressure was obtained from PLS.

2) Section 4.5.1.1.c (Pg. 3/4 5-2) Accumulators

RCS pressure of 2000 has been changed to 1000 because the Applicability Statement refers to Mode 3 above 1000 psig.

Byron station has been designed with a switch on the Main Control Board which controls the feed breakers to the SVAG valve MCC's.

3) Section 4.5.1.2 (Pg. 3/4 5-2) Accumulators

Surveillance 4.5.1.2.a has been deleted because the Byron plant design does not allow operational testing of Accumulator level while at power. Such testing would require lifting leads which injects spurious annunciator alarms in the Main Control Room.

4) Section 3/4.5.4.a (Pg. 3/4 5-9) Refueling Water Storage Tank

Decimal accuracy of 88.7% cannot be obtained from the instrument. Therefore 89.0 has been inserted.

(0423M)

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water level of between 34% (6995 gallons) and 66% (7217 gallons),

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between ~~600~~ 617 and ~~650~~ 662 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution,

c. At least once per 31 days when the SVAG valve MCC's RCS pressure is above ~~2000~~ 1000 psig by verifying that power to the ~~isolation valve operator~~ is disconnected from the circuit by ~~opening the breaker and removing the control fuses~~ verifying that SVAG valve MCC feed breaker open indication, and

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE: at

~~a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and~~

~~b.~~ ~~At~~ least once per 18 months by the performance of a CHANNEL CALIBRATION.

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LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) and the heat traced portions of the associated flow paths shall be OPERABLE with:

a. A minimum contained borated water level of 89 ~~88.7%~~ (395,000 gallons),

b. A minimum boron concentration of 2000 ppm,

c. A minimum water temperature of 35°F, and

d. A maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F.

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

Attachment H
(Section 3/4.6)

1) Section 4.6.1.1 (Pg. 3/4 6-1) Containment Integrity

   3.6.4.1.a has been changed to reflect the correct reference of 3.6.3.

2) Section 3.6.1.2.a.2 (Pg. 3/4 6-2) Containment Leakage.

   "L_t 0.07%" has been changed to "L_t 0.10%" because App J of 10 CFR 50 paragraph III.A.4 refers to a L_t of 0.10%.

3) Section 3.6.1.4 (Pg. 3/4 6-6) Internal Pressure

   There will be no further information. The internal pressure limits will remain as they are.

4) Section 4.6.1.6.2 (Pg. 3/4 6-10) End Anchorages and Adjacent Concrete Surfaces

   "Since last inspected" has been added to the end of the first main sentence for clarification. Also, "containment vessel tendon" and "4.6.1.b.1" have been added for correct references.

5) Sections 3.6.1.7.a, 4.6.1.7.1 (Pg. 3/4 6-11, 12) Containment Ventilation System

   For clarification, the phrase "sealed closed" has been changed to "power removed".

6) Section 3.6.2.2.b (Pg. 3/4 6-14) Spray Additive System

   "Chemical" has been changed to "spray" to be consistent with plant reference.

7) Table 3.6-1 (Pg. 3/4 6-19) Containment Isolation Valves

   Isolation Times for 1PS228A through 1PS230B have been deleted and a statement has been added stating. "*The proper operation of solenoid operated valves is not a time dependent function". "NA*" has replaced each Isolation Time for these valves.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification ~~3.6.4.1~~ 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 43.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.10% by weight of the containment air per 24 hours at P_a, 43.6 psig, or

2) Less than or equal to L_t, ~~0.07%~~ 0.10 by weight of the containment air per 24 hours at P_t, 21.8 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 43.6 psig, or P_t, 21.8 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

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INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and +0.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages since last inspected. Inspections of the concrete shall be performed during the ~~Type A containment leakage rate~~ Containment Vessel Tendon tests (reference Specification ~~4.6.1.2~~ 4.6.1.6.1) ~~while the containment vessel is at its maximum test pressure~~.

4.6.1.6.3 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

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CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 48-inch containment shutdown purge supply and exhaust isolation valve shall be closed and ~~sealed closed~~ Power removed, and

b. The 8-inch containment purge supply and exhaust isolation valve(s) may be open for up to 1000 hours during a calendar year provided no more than one line is open at one time.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve open or not sealed closed, close and/or seal close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 8-inch containment purge supply and/or exhaust isolation valve(s) open for more than 1000 hours during a calendar year, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.3 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve(s) shall be verified ~~sealed~~ closed and ~~closed~~ power removed at least once per 31 days.

4.6.1.7.2 The cumulative time that all 8-inch containment purge supply and/or exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS, the inboard and outboard valves with resilient material seals in each sealed closed 48-inch containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to at least P_a, 43.6 psig.

4.6.1.7.4 At least once per 3 months, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to at least P_a, 43.6 psig.

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SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a level of between 78.6% (4000 gallons) and 90.3% (4540 gallons) of between 30 and 36% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the ~~chemical~~ spray additive tank to a Containment Spray System pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Spray Actuation test signal; and

d. At least once per 5 years by verifying each water flow rate equivalent to 55(+5,-0) gallons per minute for 30% NaOH from the Educator test connections in the Spray Additive System:

1) CS26A $68 \begin{smallmatrix}+6\\-0\end{smallmatrix}$ gpm (Train A), and

2) CS26B $68 \begin{smallmatrix}+6\\-0\end{smallmatrix}$ gpm (Train B).

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (s)	TYPE OF OPERATOR
1. Phase "A" Isolation (Continued)			
1PS228A	Process Sampling	~~0.10~~ NA*	Solenoid
1PS229A	Process Sampling	~~0.10~~ NA*	Solenoid
1PS230A	Process Sampling	~~0.12~~ NA*	Solenoid
1PS228B	Process Sampling	~~0.10~~ NA*	Solenoid
1PS229B	Process Sampling	~~0.10~~ NA*	Solenoid
1PS230B	Process Sampling	~~0.12~~ NA*	Solenoid
1PS9354A	Process Sampling	10	Air Operator with solenoid accessory
1PS9354B	Process Sampling	10	Air Operator with solenoid accessory
1PS9355A	Process Sampling	10	Air Operator with solenoid accessory
1PS9355B	Process Sampling	10	Air Operator with solenoid accessory
1PS9356A	Process Sampling	10	Air Operator with solenoid accessory
1PS9356B	Process Sampling	10	Air Operator with solenoid accessory
1PS9357A	Process Sampling	10	Air Operator with solenoid accessory
1PS9357B	Process Sampling	10	Air Operator with solenoid accessory
1RE9157	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9159A	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9159B	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9160A	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9160B	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE1003	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RE9170	Reactor and Containment Drains to Radiowaste	10	Air Operator with solenoid accessory
1RY8025	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory
1RY8026	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory
1RY8033	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory
1RY8028	Reactor Coolant Pressurizer	10	Air Operator with solenoid accessory

* Proper valve operation will be demonstrated by verifying the valve strokes to ~~their~~ its required position.

Attachment I
(Section 3/4.7)

1) Section 3.7.1.2 (Pg. 3/4 7-4) Auxiliary Feedwater System

The day tank level instrument reads in precentage; "420 gallons" is replaced by "level of 71%" in section 3.7.1.2.b, page 3/4 7-4.

The suggested change in Action Statement 3.7.1.2.a to "feedwater pump" from "feedwater pumps" is necessary because Byron Station has only two pumps.

2) Section 3.7.4 (Pg. 3/4 7-12) Essential Service Water System

Statement 3.7.4.b has been deleted from page 3/4 7-12 and moved to Statement 3.7.5.d so statement 3.7.4 will agree with FSAR Amendment 36 Q10.48-1 (A copy of FSAR Amendment 36 Q10.48-1 has been attached). The word "Makeup" has been added todescribe the pumps.

3) Section 4.7.6.c.1, 4.7.6.c.3, 4.7.6.e.1, 4.7.6.f and 4.7.6.8 (Pg. 3/4 7-16, 17) Control Room Emergency Air Cleanup System

Since Make-up is enough to provide pressurization, no credit is taken for the normal supply system. The Emergency cleanup system consists of only the make-up filter units. Statements 4.7.6.c.1, 4.7.6.c.3, 4.7.6.e.1, 4.7.6.f, and 4.7.6.g have changed "Pressurization System" to "Make-up System" and deleted "and 51,000 cfm $\pm$ 10% for the Recirculation System." In 4.7.6.c.2, iodine is changed to iodide because Methyl Iodide is the correct form of the compound.

4) Section 4.7.6.3.2 (Pg 3/4 7-16) Control Room Emergency Air Cleanup System

The "Smoke Density-High" of 4.7.6.e.1 has been changed to "Safety Injection" because the "Smoke Density-High" is only an ionization detection system that routes the supply air through a charcoal adsorber bank prior to entering the supply fan. No credit is taken for radiological protection or particulate.

5) Section 4.7.7.b.2 (Pg. 3/4 7-18) Non-Accessible Area Exhaust Filter Plenum Ventilation System

"Iodine" is changed to "iodide" because Methyl Iodide is the correct form of the compound.

6) Section 3.7.9.2.c (Pg. 3/4 7-27) Sealed Source Contamination

Since primary startup sources are too radioactive to be handled for source checks, a note has been referenced to the word Startup Sources which states "*Except Primary Startup Sources used for initial cycle Startup". A "*" has been added after the word "sources" in Section 3.7.9.2.c.

Attachment I, (Continued)
(Section 3/4.7)

7) Section 3.7.10.2.c.2 (Pg. 3/4 7-31) Foam Systems

For operating convenience, "Spray and sprinkler" has been changed to "deluge" to verify the integrity of dry pipe.

8) Section 4.7.10.3.2.a (Pg. 3/4 7-32) CO_2 Systems

Surveillance 4.7.10.3.2.a has been replaced in order to agree with the design of Byron Station.

With the statement "at least once per 7 days by verifying the plant CO_2 storage tank level to be greater than 75% (7.5 tons), the river screen house CO_2 storage tank level to be greater than 50% (1 ton), and pressure of both to be greater than 290 psig, and"

and less than [illegible]

9) Section 3.7.10.5.a (Pg. 3/4 7-34) Fire Hose Stations

The term "operating technicians" is replaced by "operating personnel" to reflect Byron Station terminology.

10) Section 3/4.7.11.2 (Pg. 3/4 7-38) Fire Rated Assemblies

Delete the phrase "automatic hold-open, release and" in Section 4.7.11.2 because the station fire doors do not have automatic hold-open, release mechanisms.

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least two independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. One motor-driven auxiliary feedwater pump capable of being powered from an ESF Bus, and

b. One direct-driven diesel auxiliary feedwater pump capable of being powered from a direct-drive diesel engine and an OPERABLE Diesel Fuel Supply System consisting of a day tank containing a minimum ~~of~~ level of 71% (420 gallons) of fuel.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater ~~pumps~~ pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. X

b. With both auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that the pump develops a differential pressure of greater than or equal to 1825 psid at a flow of greater than or equal to 85 gpm in the recirculation mode;

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3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent Essential Service Water Systems, which includes a loop and a cooling tower, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only one Essential Service Water System OPERABLE, restore at least two Essential Service Water Systems to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a loss-of-offsite power and with the outside air temperature equal to or less than 40°F, start both essential service water makeup pumps unless already operating.

Move to next page

SURVEILLANCE REQUIREMENTS

4.7.4 At least two Essential Service Water Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment or isolating the non-nuclear safety-related portion of the system actuates to its correct position on a Safety Injection test signal, and

2) Each Essential Service Water System pump starts automatically on a Safety Injection test signal.

c. At least once per 31 days, by verifying that each cooling tower fan operates for at least 15 minutes and at least once per 18 months by visually inspecting and verifying no abnormal breakage or degradation of the fill materials in the cooling tower.

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3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with:

a. An essential service water pump discharge water temperature of less than or equal to 98°F, and

b. 1) A minimum Rock River water level at or above 665 Mean Sea Level, USGS datum, at the Byron Screenhouse, with two essential service water make-up pumps OPERABLE, or

2) Two deep wells OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the essential service water pump discharge water temperature not meeting the above requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the minimum Rock River water level not meeting the above requirement and one deep well inoperable, restore both deep wells to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one essential service water make-up pump inoperable and one deep well inoperable restore both essential service water make-up pumps or both deep wells to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Add Action Statement b from Page 3/4 7-12

d.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The UHS shall be determined OPERABLE at least once per 24 hours by verifying the essential service water pump discharge water temperature and the Rock River water level to be within their limits.

4.7.5.2 The deep wells shall be demonstrated OPERABLE:

a. At least once per 31 days by starting each pumps and operating it for at least 15 minutes and verifying that each valve (manual power-operated, or automatic) in the flow path is in its correct position, and

b. At least once per 18 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

QUESTION 010.48

"Provide an analysis of the minimum temperature conditions which will be reached in the Byron river screen house following prolonged loss of the building unit heaters or loss of offsite power during extreme cold weather. Define the minimum operating temperature conditions at the essential service water makeup pump diesel drive units, the diesel oil supply system, and the essential service water lines as a function of time from heating system failure and of ambient temperature. State the reliability of starting the diesel drive units and of provisions to prevent freezing in stagnant water lines during the minimum temperature period."

RESPONSE

An analysis was performed to determine the minimum temperature conditions which will be reached in the Byron river screen house following a prolonged loss of the building heaters during extremely cold weather due to loss of power. The analysis is based on ambient conditions of -10° F, a 15 mph windspeed, and a 65° F inside temperature. The results of the analysis show that the river screen house will reach a temperature of 40° F in approximately 30 minutes. This is a sufficient amount of time for plant personnel to be sent to the screen house and start the diesels (loss of power at the RSH is annunciated in the control room by the telemetry system powered by a backup DC battery). The diesels will be qualified to start at 40° F with no loss in reliability.

Byron station procedures will specify starting the essential service water makeup pumps upon a River Screen House HVAC trouble annunciator coincident with ambient temperatures below 40° F.

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c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10% for the ~~Pressurization~~ Make-up System, ~~and 51,000 cfm ± 10% for the Recirculation System;~~

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl ~~iodine~~ iodide penetration of less than 0.2%; and X

3) Verifying a system flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ Make-up System ~~and 51,000 cfm ± 10% for the Recirculation System~~ when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodine penetration of less than 0.2%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.0 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ Make-up System. ~~and 51,000 cfm ± 10% for the Recirculation System,~~

2) Verifying that on a ~~Smoke Density-High~~ Safety Injection or High Radiation-Control Room (Outside Air Intake, or Turbine Building Intake) test signal, the system automatically switches into a ~~recirculation~~ Make-up mode of operation with flow through the HEPA filters and charcoal adsorber banks,

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the control room at a positive nominal pressure of greater than or equal to 1/8 inch Water Gauge relative to ambient pressure in areas served by the system, and

4) Verifying that the heaters dissipate 27.2 ± 2.7 kW when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ Make-up System. ~~and 51,000 cfm ± 10% for the Recirculation System; and~~

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 6000 cfm ± 10% for the ~~Pressurization~~ Make-up System. ~~and 51,000 cfm ± 10% for the Recirculation System.~~

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3/4.7.7 NON-ACCESSIBLE AREA EXHAUST FILTER PLENUM VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Three independent Non-Accessible Area Exhaust Filter Plenum Ventilation Systems (50% capacity each) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Non-Accessible Area Exhaust Filter Plenum Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Non-Accessible Area Exhaust Filter Plenum Ventilation System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

1) Verifying that with the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 66,900 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl ~~iodine~~ iodide penetration of less than 7.1%; X

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SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources* and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

* Except Primary Startup sources used for initial cycle startup

(A) At least once per 7 days by verifying the plant CO_2 storage tank level to be greater than 75% (7.5 tons), the river screen house CO_2 storage tank level to be greater than 50% (1 ton), and pressure of both to be greater than 290 and less than 375 psig, and

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.5 The fire hose stations given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-5 inoperable, provide gated wye (s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating personnel ~~technicians~~, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the operable hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.5 Each of the fire hose stations given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station,

b. At least once per 18 months, by:

    1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

    2) Removing the hose for inspection and re-racking, and

    3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

    1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

    2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

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3/4.7.11 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.11 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire-rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire barrier penetrations and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware,

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the ~~automatic hold-open,~~ release and closing mechanism and latches at least once per 6 months, and by verifying: X

a. The OPERABILITY of the Fire Door Supervision System for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door is closed at least once per 7 days, and

c. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

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FOAM SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.2 The Foam Systems in the diesel generator fuel storage tank rooms shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the Foam System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Foam Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Foam Systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a Fire Detection test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2) By a visual inspection of the dry pipe deluge ~~spray and sprinkler~~ headers to verify their integrity, and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open deluge nozzle and verifying each nozzle is unobstructed.

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PLANT SYSTEMS

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The following Low Pressure CO_2 Systems shall be OPERABLE:

a. Diesel generator rooms and day tank rooms,

b. Lower cable spreading room,

c. Auxiliary feedwater diesel room and day tank room, and

d. Diesel-driven ESW make-up pumps and day tank rooms.

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.3.1 Each of the above required Low Pressure CO_2 Systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

4.7.10.3.2 Each of the above required Low Pressure CO_2 Systems shall be demonstrated OPERABLE:

Replace with (A)

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 75% (7.5 tons) and pressure to be greater than 290 psig, and

b. At least once per 18 months by verifying:

1) The system, including associated ventilation system fire dampers, actuates manually and automatically upon receipt of a simulated actuation signal, and

2) Flow from each nozzle during a "Puff Test."

Attachment J
(Section 3/4.8)

1) Section 4.8.1.1.1 (Pg. 3/4 8-4)) A.C. Sources

   Deletion of "SX Pump" is requested to allow the station to meet the intent of this surveillance in the event that the SX pump is unavailable. The reason is explained in the Basis.

2) Section 3.8.3.1 (Pg. 3/4 8-14) Onsite Power Distribution Operating

   For consistency in Technical Specification, "4160-volt" has been changed to "4kv".

3) Section 4.8.42 (Pg. 3/4 8-34) Motor-operated valves thermal Overload Protective Device)

   "Normally in force" has been deleted because devices will always be in force.

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A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries is based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.05 volts, ensures the battery's capability to perform its design function.

The station chose its largest emergency load to be the SX pump.
The maximum BHP of the SX pump is 1249 per FSAR Table 8.3-1
A BHP of 1249 correspond to a load of 1034 kw

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4) An impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70, analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and

5) The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977, analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.

e. At least once per 18 months, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2) Verifying the generator capability to reject a load of greater than or equal to 1034 kW ~~(SX Pump)~~ while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 4.5 Hz,

3) Verifying the diesel generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection,

4) Simulating a loss of ESF bus voltage by itself, and:

a) Verifying de-energization of the ESF busses and load shedding from the ESF busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the ESF busses with permanently connected loads within 10 seconds, energizes the auto-connected safe shutdown loads through the load sequencing timer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 420 volts and 60 ± 4.5 Hz during this test.

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3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open (both) between redundant busses within the unit (and between units at the same station):

a. Division 11 A.C. ESF Busses consisting of:
   1) ~~4160-Volt~~ 4kV Bus 141,
   2) 480-Volt Bus 131X, and
   3) 480-Volt Bus 131Z.

b. Division 12 A.C. ESF Busses consisting of:
   1) ~~4160-Volt~~ 4kV Bus 142
   2) 480-Volt Bus 132X, and
   3) 480-Volt Bus 132Z.

c. 120-Volt A.C. Bus 111 energized from its associated inverter connected to D.C. Bus 111,*

d. 120-Volt A.C. Bus 113 energized from its associated inverter connected to D.C. Bus 111,*

e. 120-Volt A.C. Bus 112 energized from its associated inverter connected to D.C. Bus 112,* and

f. 120-Volt A.C. Bus 114 energized from its associated inverter connected to D.C. Bus 112.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. ESF busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus not energized, reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. inverter inoperable or not connected to its D.C. power supply, reenergize the A.C. vital bus from its associated inverter within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

*Two inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

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MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection devices, integral with the motor starter of each valve listed in Table 3.8-2, shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection devices shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

a. All thermal overload devices, such that each device is calibrated at least once per 6 years, and

b. All thermal overload devices ~~normally in force~~ such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE, at least once per 6 years.

Attachment K
(Section 3/4.9)

1) Section 3.9.1 (Pg. 3/4 9-1) Boron Concentration

"By" has been added to LCO statement for clarification.

2) Section 4.9.1.1 (Pg. 3/4 9-1) Boron Concentration

"Full length" is not applicable to Byron Station and was thus deleted because there is no need for rod length designation.

3) Section 3.9.3 (Pg. 3/4 9-3) Decay Time

"During" has been changed to "prior to" to be consistent with Basis 3/4.9.3 statement.

4) Section 3.9.4 (Pg. 3/4 9-4) Containment Building Penetrations

"Personnel" has been changed to "equipment". Equipment is the station approved terminology. In order to eliminate present confusion at the rate, "removed or" has been rearranged.

5) Section 4.9.7 (Pg. 3/4.9.7) Crane Travel

The suggested change to 2000 from 2250 is based on NREG 068.

(0423M)

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met either:

by

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any ~~full-length~~ control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Valves 1CV8455, 1CV8464A, 1PW046, and 2PW046 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: Prior to ~~During~~ movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The ~~personnel~~ equipment hatch removed or closed and held in place by a minimum of four bolts, ~~or removed,~~

~~b. A minimum of one door in the personnel hatch is closed,~~

b. A minimum of one door in the personnel emergency exit hatch is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Be capable of being closed by an OPERABLE automatic containment purge isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge isolation valves per the applicable portions of Specification 4.6.3.2.

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3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility.

APPLICABILITY: With fuel assemblies in the spent fuel storage facility.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of ~~2250~~ 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

Attachment L
(Section 3/4.10)

1) Section 3/4 10.1 (Pg. 3/4 10-1) Shutdown Margin

"Full length" is not applicable to Byron Station and was thus deleted in sections 3.10.1.a, 3.10.1.b, 4.10.1.1., and 4.10.1.2. The phrase "or equivalent boron rate" has been added for clarification.

2) Section 3/4 10.4 (Pg. 3/4 10-4) Reactor Coolant Loops

Reference to "STARTUP" has been deleted. According to FSAR Section 14.2.3, the term "STARTUP TEST" is reserved for initial plant testing after core load and prior to initial 100% power operation. All testing of the reactor core is referred to as "Physics Testing".

3) Section 4.10.5 (Pg. 3/4 10-5) Position Indiction System - Shutdown

The phrase "the start of" has been changed to "startup" for clarification.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any ~~full-length~~ control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm ~~of~~ a solution containing greater than or equal to 7000 ppm boron ~~or its equivalent~~ until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

or equivalent boration rate

b. With all ~~full-length~~ control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron ~~or its equivalent~~ until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each ~~full-length~~ control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each ~~full-length~~ control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of ~~STARTUP and~~ PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during ~~STARTUP and~~ PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating ~~STARTUP and~~ PHYSICS TESTS.

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3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The digital rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to ~~the~~ start ~~of~~ up and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree: α

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of the Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

Attachment M
(Section 3/4.11)

1) Table 4.11-1 (Pg. 3/4 11-4) Table Notations

"S-1" is replaced by "Sec^{-1}" for clarity in the definition of lambda.

"S" is replaced by "Sec" for clarity in the definition of delta t.

2) Table 4.11-1 (Pg. 3/4 11-5) Table Notations

To indicate and describe Radioactivity Monitors providing alarm but not providing Automatic Termination of Release, the notation statement (7) reads "Essential Service Water RCFC Radiation Oulet Monitor (1RE-PR002) and (1RE-PR003)".

3) 3.11.1.2 (Pg. 3/4 11-6) Radioactive Effluents Dose

Delete "This Special Report shall also include: (1)...... Safe Drinking Water Act.*" and "*The requirements of Action a.(1)...... 3 miles downstream only". These items are no longer needed because the Rock River is not used for drinking water within 3 miles downstream of the plant discharge.

4) Section 3/4.11.2 (Pg. 3/4 11-9) Dose Rate

For statements 3.11.2.1, 4.11.2.1.1, and 4.11.2.1.2, the word "rate" in "dose rate" is deleted. (5 places). For gases the term dose is used, not dose rate.

5) Table 4.11-2 (Pg 3/4 11-11) Table Notations

In the last paragraph, the phrases "a prior" and "a posterior"; are deleted because of redundancy. Paranthesis in this paragraph are removed and the word "an" becomes "a" for clarity and proper grammer.

6) Table 4.11-2 (Pg. 3/4 11-12) Table Notations (5)

"Ventilation exhaust from the" is deleted from item 5 because Tritium Samples will be from drains on area dehumidifier (spent fuel pit area) which leads to ventilation exhaust.

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Sec^{-1}

Δt = the elapsed time between the midpoint of sample collection and the time of counting (~~s~~). sec

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(3) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. 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 6.7.1.7 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

RCFC Outlet

(7) Not required unless the Essential Service Water ~~Outlet Line~~ Radiation Monitor (1RE-PR002) indicates measured levels greater than 1 x 10^{-6} μCi/ml above background at any time during the week.

and (1RE-PR003)

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.7.2, a Special Report that identifies the cause(s) for exceeding the limit(s) 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. ~~This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Safe Drinking Water Act.*~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

~~*The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river-sited plants this is 3 miles downstream only.~~

RADIOACTIVE EFFLUENTS

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3/4.11.2 GASEOUS EFFLUENTS

DOSE ~~RATE~~

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose ~~rate~~ due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131 and 133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose(s) ~~rate(s~~) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose ~~rate~~ due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose ~~rate~~ due to Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a~~n~~ ~~a priori~~ (before the fact) limit representing the capability of a measurement system and not as an ~~a posteriori~~ (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. 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 6.7.1.7, in the format outlined in Regulator Guide 1.21, Appendix B, Revision 1, June 1974.

(3) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5) Tritium grab samples shall be taken at least once per 7 days from the ~~ventilation exhaust from the~~ spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

Attachment N
(Section 3/4 12)

1) Section 3/4 12.1 (Pg. 3/4 12-2) Monitoring Program

Submission of controlled copies of the ODCM is an acceptable alternative to submission of the required information via the Semiannual Radioactive Effluent Report. This method was discussed and agreed upon in phone conversations between John Golden of Commonwealth Edison and NRC officials in Bethesda.

2) Table 3.12-1 (Pg. 3/4 12-3) Radiological Environmental Monitoring Program

Commonwealth Edison does not site direct radiation monitors at nearby residences and schools because of the enhanced loss factor using these locations. We do site TLDs in an inner ring and outer ring as noted in the Technical Specifications and at population centers where there are air sampling facilities. The security aspect is important because without it one cannot have reasonable assurance that the TLDs, which is the direct radiation monitoring device used by Commonwealth Edison, will be present for the specified period of time on a continuing basis. Locating the devices at schools and residences increases by a considerable amount the likelihood of vandalism.

3) Table 3.12-1 (Pg. 3/4 12-4) Airborne Monitoring

The range to the Commonwealth Edison Byron Station control location is approximately 10 kilometers. These monitoring sites are located in Mount Morris and Leaf River to the west and northwest of the plant. Both locations being in the least prevalent wind direction. The approximate distance is seven-eight miles which is more than adequate to establish a control for the site. Four onsite air-monitoring sites will be added. In addition, given that Commonwealth Edison has some 60 or so air sampling stations throughout northern Illinois associated with it's nuclear program, there are ample control locations elsewhere if necessary for any particular analysis.

4) Table 3.12-1 (Pg. 3/4 12-4) Waterborne Sampling 3a

The suggested change in language to weekly collection composited monthly is proposed because there is no public water intake on the Rock River. Furthermore, there is a composite sampling device on the discharge water line carrying the liquid waste to the receiving body of water.

5) Table 3.12-1 (Pg. 3/4 12-4) Waterborne Monitoring 3b

There are not groundwater sources likely to be affected by plant operation. However, Commonwealth Edison does monitor the onsite well water source used by the plant and one nearby offsite source for purposes of confirmatory surveillance and will do so in this program.

6) Table 3.12-1 (Pg. 3/6 12-4) Waterborne Monitoring 3c

There is no public water supply using the Rock River, hence, this section is not applicable to the Byron Environmental Monitoring Program.

7) Table 3.12-1 (Pg. 3/4 12-5) Milk 4a

The suggested change to samples from three nearby dairies, if possible the nearest ones, is believed necessary in order to provide a quality Environmental Monitoring Program and to assure continuity of records and collection of a representative sample. We do not monitor so called "family milking animals" because to do so would require a week-by-week census in order to determine whether or not they are present and also because these animals are usually there to supply milk only for family consumption and is thus not available for sale to others. However, should there be an accident at Byron Station involving releases of significant quantities of radioactive material, every effort will be made to monitor all nearby animals. Both those at dairies as well as the family animals.

8) Table 3.12-1 (Pg. 3/6 12-5) Fish and Invertebrates 4b

The suggested change to representative sample of commercially and recreationally important species is believed necessary because it is impossible to document that one has collected one sample of each commercially and recreationally important species. A suggested change in the sampling frequency to three times per year (spring, summer & fall) is made to put the monitoring program on a more consistent and documentable schedule than merely sampling in season or semiannually, if they are not seasonal. This aspect of the requirements would be extremely difficult to document.

9) Table 3.12-1 (Pg. 3/4 12-5) Food Product 4c

The suggested change in the representative sample of each principal class of food products grown within 10 miles of the plant is believed necessary because irrigation suing water from the Rock River occurs throughout the entire river and it would be difficult, if not impossible, to get a sample of each principal class of food products grown for over a 100 miles. Samples are currently attained at commercial stands downriver of the plant discharge and are considered to be representative of those foods grown in the vicinity of the plant.

10) Table 3.12-1 (Pg. 3/4 12-7) Table Notations (1)

The second sentence beginning with "Refer to NUREG-013, etc.", was deleted because it has no relevancy to the rest of the paragraph. Also in this same footnote is a reference to reporting in the Semi-Annual Radioactive Effluent Release Report which was discussed previously in Item. 1.

11) Table 3.12-1 (Pg. 3/4 12-7) Table Notations (5)

Delete the two last sentences because they do not apply to Byron Station.

12) Table 3.12-1 (Pg. 3/4 12-8) Table Notations (6)

This footnote was deleted because it is no longer applicable as a result of discussion in Item 4 above.

13) Table 3.12-1 (Pg. 3/4 12-8) Table Notations (7)

Is not applicable and was thus deleted because there is no groundwater source suitable for contamination.

14) Table 3.12-1 (Pg. 3/4 12-8) Table Notations (8)

This footnote was deleted because its source of reference has been deleted.

15) Table 4.12-1 (Pg. 3/4 12-10) Detection Capabilities for Environmental Sample Analysis

The suggested change from 0.07 to 0.10 is based on Quality Assurance of Teledyne isotope.

16) Section 3/4 12.2 (Pg. 3/4 12-13) Land Use Census

The language concerning garden census was deleted because there is no groundwater source suitable for contamination.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.7.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.7.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report required by Specification 6.7.1.6.

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. ~~Pursuant to Specification 6.12, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the~~ new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

Add

Delete

Submit controlled version of the ODCM within 180 days including a revised figure(s) and table reflecting the

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation[2]	Forty routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site; and The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations. ← Delete at the air sampling sites.	Quarterly.	Gamma dose quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations: Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground level D/Q; One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q; and One sample from a control location, as for example ~~15~~ 10 to 30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis ~~weekly~~. every two weeks Particulate Sampler: Gross beta radioactivity analysis following filter change;(3) and gamma isotopic analysis(4) of composite (by location) quarterly.
3. Waterborne			
a. Surface(5)	One sample upstream. One sample downstream.	Delete → weekly collection composited monthly → ~~Composite sample over 1-month period.(6)~~	Gamma isotopic analysis(4) monthly. Composite for tritium analysis quarterly.
b. Ground	~~Samples from one or two sources only if likely to be affected(7).~~ Samples from one onsite and one offsite source	Quarterly.	Gamma isotopic(4) and tritium analysis quarterly.
~~c. Drinking~~	One sample of each of one to three of the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period(6) when I-131 analysis is performed, monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.(8) Composite for gross beta and gamma isotopic analyses(4) monthly. Composite for tritium analysis quarterly.

DELETE

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)			
C ~~d~~. Sediment from shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis(4) semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 5 km distance having the highest dose potential. If there are none, then, one sample from milking animals in each of three areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr(8). Delete Samples from three dairies within 8 km, the nearest one to the plant, if possible	Semimonthly when animals are on pasture, monthly at other times.	Gamma isotopic(4) and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
	One sample from milking animals at a control location, 15 to 30 km distant and in the least prevalent wind direction.	Three times per year (Spring, summer, and fall)	
b. Fish and Invertebrates	Representative Samples of ~~One sample of each~~ commercially and recreationally important species in vicinity of plant discharge area. ~~One sample of same species in areas not influenced by plant discharge.~~	~~Sample in season, or semiannually if they are not seasonal.~~ Representative samples of commercially and recreationally important species in areas not influenced by plant discharge	Gamma isotopic analysis(4) on edible portions.
c. Food Products	~~One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.~~ Representative samples of the principal classes of food products from any area within 10 miles of the plant that is irrigated by water in which liquid plant wastes have been discharged	At time of harvest.(~~9~~ 6)	Gamma isotopic analyses(4) on edible portion.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
c. Food Products (continued)	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic(4) and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic(4) and I-131 analysis.

Submit controlled revisions of the ODCM within 180 days including a revised figure(s) and table

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from the centerline of one unit, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. ~~Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979.~~ Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavilability, malfuntion of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.7.1.6. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. ~~Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM~~ reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

(2) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

(3) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(5) The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

(6) A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

Delete

(7) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(8) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

6(~~9~~) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/ℓ may be used.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[1]

LOWER LIMIT OF DECTECTION (LLD)[2][3]

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GAS (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[4]	0.10 ~~0.07~~		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/ℓ may be used.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.7.1.6.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yeild a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide, (s^{-1}), and

Δt = the elapsed time between sample collection, or end of the sample collection period, and time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.7.1.6.

(4) LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

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3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, ~~the nearest residence and the nearest garden* of greater than 50 m² (500 ft²) producing broad leaf vegetation~~. and the nearest residence

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next ~~Semiannual Radioactive Effluent Release Report~~, pursuant to Specification 6.7.1.~~7~~ 6.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. Pursuant to Specification ~~6.14~~ 6.7.1.6, submit in the next ~~Semiannual Radioactive Effluent Release Report~~ documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

Annual Radiological Environmental Operating Report

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to. Specification 6.7.1.6.

~~*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4c. shall be followed, including analysis of control samples.~~

Delete

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials, suppled as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.7.1.6.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.7.1.6.

Attachment O
(3 & 4 Bases)

1) Basis Section 3/4.1.1.1 and 3/4.1.1.2 (Pg. B 3/4 1-1)

   Insertain is a typographical error. The correct word should be "insertion".

2) Basis Section 3/4 1.3 (Pg. B 3/4 1-4)

   Add the alarm to description: automatic monitoring channel, "alarm".

3) Section 3/4.2.2.a (Pg. B 3/4 2-2)

   Replace the word "insertion" with "position". "Position" more accurately describes the action.

4) Basis Section 3/4.2.2 (Pg. B 3/4 2-5)

   Delete "only" in last paragraph. "Only" limits the surveillance as to its accomplishments.

5) Basis Section 3/4.3.3.1 (Pg. B 3/4 3-3)

   Typographical Error; "actuation" was mispelled acutation.

6) Basis Section 3/4.4.5 (Pg. 13 3/4 4-3)

   Specification 6.7.1 should be changed to collect specification "6.7.2".

7) Basis Section 3/4.4.8 (Pg. B 3/4 4-5)

   The following typographical errors are corrected: "E" becomes "$\bar{E}$", "cooland" becomes "coolant" and "trasport" becomes "transport". For clarity a comma is added after "rupture" and the phrase "at this temperature" is added after coolant in the last paragraph on page B 3/4 4-6.

   Delete the word probably to add decisiveness to statement.

8) Basis Section 3/4.4.9 (Pg. B 3/4 4-8)

   Add a comma and delete "and" after the word "fluence" to make the sentence grammerically correct.

Attachment O (Continued)
(3 & 4 Bases)

9) Basis Section 3/4.6.1.6 (Pg. B 3/4 6-2) Containment Vessel Structural Integrity

Update paragraph two to read: The surveillance requirements for demonstrating the containments structural integrity are in compliance with the recommendations of proposed Rev. 3 to Regulatory Guide 1.35 "Inservice Surveillance of ungrouted tendous in Prestressed Concrete Containment Structures," April 1979 and proposed Regulatory Guide 1.35.1 "Determining Prestressing Forces for Inspection of Prestressed concrete containments" April 1979. This updates the paragraph from 1976 to 1979.

10) Basis Section 3/4.6.2.2 (Pg. B 3/4 6-3)

"Evolution of Iodine" should read "evolution of Hydrogen" because band is also based on maintaining Iodine in solution.

11) Basis Section 3/4.7.1.1 (Pg. B 3/4 7-1)

The phrase steam "bypass" to the condenser should read steam "dumps" to the condenser to give correct terminology.

12) Basis Section 3/4.3.1.4 (pg. B 3/4 7-2)

Replace "rupture" with "break" to include all steamline breaks.

13) Basis section 3/4.8.1-3 (Pg. B 3/4 8-2) A. C. Sources, D. C. Sourses, and Onsite Power Distribution

Deletion of "SX Pump" from section 4.8.1.1.1 "SX Pump" was requested to allow the station to meet the intent of the surveillance section 4.8.1.1.1 in the event that the SX pump is unavailable. The reason is explained in the Basis on page B 3/4 8.2.

14) Basis Section 3/4.9.4 (Pg. B 3/4 9-1)

Replace "in" with "from" because radioactivity escapes from the containment building, not in it.

15) Basis Section 3/4.10.1 (Pg. 13 3/4 10-1)

Add " and shutdown margin" before the work "measurement" in the first sentence. Test are performed for both rodworth and shutdown margins.

16) Basis Section 3/4.10.2 (Pg. B 3/4 10-1)

The word "damping" is correctly spelled "dampening".

17) Basis Section 3/4.11.1.2 (Pg. B 3/4 11-1)

Delete sentence referring to drinking water. The river is not a source of drinking water.

18) Basis Section 3/4 11.2.2 (Pg. B 3/4 11-4)

The word "liquid" should be replaced by "gas" because this section refers to noble gases.

The word "models" is typed as "modesl".

19) Basis Section 3/4 11.2.3 (pg. B 3/4 11-5)

Change the word "of" to "to" in the second last word of the first paragraph. The change adds clarity to the sentence.

20) Basis Section 3/4 11.2.4 (Pg. B 3/4 11-6)

Replace "Liquid" with "Gaseous" for conformity with the section title.

21) Basis Section 3/4 12.1 (Pg. B 3/4 12-1)

Delete "a priori", "a posteriori" the parenthesis around "before the fact" and "after the fact" and change "an" to "a". This eliminates redundancy.

22) Basis 3/4 12.2 (Pg. B 3/4 12-1) Land Use Census

The language concerning garden census was deleted because Commonwealth Edison assumes a garden exists at each nearest residence. Since the purpose of the census is to locate those gardens appropriate for pathway analysis is the ODCM the assumption that a garden exists at each point, negates the need for the census. This also applies to the foot note on the bottom of page 3/4 12-13 concerning the census and also applies to the census discussion in Section 3/4.12.2.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

The OPERABILITY of the four charging pump suction valves ensures adequate capability for negative reactivity insert~~ain~~ion to prevent a transient caused by the uncontrolled dilution of the RCS. The functioning of the valves precludes the necessity of operator action to prevent further dilution by terminating flow to the charging pumps from possible unborated water sources and initiating flow from the RWST. Actions taken by the microprocessor if the neutron count rate is doubled will prevent return to criticality in these MODES.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MOVABLE CONTROL ASSEMBLIES (Continued)

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 550°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

~~alarms~~
alarm

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AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod position ~~insertion~~ differing by more than $\pm$ 12 steps, indicated, from the group demand position,

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.7.1.9 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the results to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect [redacted] flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does

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Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block of Safety Injection low pressurizer pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 An increasing steam generator water level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated ACTION will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

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3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.7.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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SPECIFIC ACTIVITY (Continued)

rupture. The reporting of cumulative operating time over 500 hours in any 6-month consecutive period with greater than 1 microCurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would ~~probably~~ be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport and analyze. The second consideration is the predictable delay time between the postualted release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radio-chemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the reactor coolant, at this temperature, is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

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PRESSURE/TEMPERATURE LIMITS (Continued)

properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, and copper content and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 43.9 psig in the event of a cold leg double-ended break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's proposed Rev 3 to structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," ~~January 1976~~ April 1979 and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalties shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, the results of the engineering evaluation and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 48-inch containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

CONTAINMENT VENTILATION SYSTEM (Continued)

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation with one line open will be limited to 1000 hours during a calendar year. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to support the additional time requests.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of ~~iodine~~ hydrogen and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam ~~bypass~~ dumps to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17.958×10^6 lbs/h which is 119% of the total secondary steam flow of 15.135×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109).$$

For three loop operation:

$$SP = \frac{(X) - (Y)(U)}{X} \times (*).$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

U = Maximum number of inoperable safety valves per operating steam line,

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SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. This value left blank pending NRC approval of three loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. The diesel-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1450 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line ~~rupture.~~ break. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

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A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries is based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.05 volts, ensures the battery's capability to perform its design function.

The station chose its largest emergency load to be the SX pump. The maximum BHP of the SX pump is 1249 per FSAR Table 8.3-1. A BHP of 1249 correspond to a load of 1034 kW.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on Keff of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations and includes a 1% Δk/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The Byron Station is designed such that the containment opens into the fuel building through the personnel hatch. In the event of a fuel drop accident in the containment, any gaseous radioactivity escaping ~~to~~ from the containment building will be filtered through the Fuel Handling Building Exhaust Ventilation System.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

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3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

and shutdown margin

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and ~~damping~~ dampening factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 10% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM-SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time is derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." ~~Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141.~~ The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides

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DOSE RATE (Continued)

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in ~~liquid~~ gas effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on ~~modest~~ models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1," July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

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3/4.11.2.3 DOSE - IODINE-131 AND 133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure ~~of~~ to man.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

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3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the ~~Liquid~~ Gaseous Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS DECAY TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity that provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem, the annual dose limit in 10 CFR Part 20.

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3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a ~~a priori~~ (before the fact) limit representing the capability of a measurement system and not as an ~~a posteriori~~ (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identfied and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegatbles assumed in Regulatory Guide 1.109 for consumption by a child. To

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BASES

LAND USE CENSUS (Continued)

determine this minimum gardent size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

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The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.